4. Policy Considerations and Environmental Impacts

This Chapter of the Environmental Impact Statement (EIS) describes the policy considerations and potential environmental impacts resulting from each of the management alternatives for implementation of the proposed action and the No Action Alternative. The environmental analysis addresses potential impacts of each alternative on workers, the public, and the environment. The general methodology used throughout this chapter is discussed in Section 4.1.

The policy considerations and environmental impacts of policy alternatives are described in this chapter. One policy alternative is the proposed action, which proposes the adoption of a policy whereby the United States would become involved in the management of the foreign research reactor spent nuclear fuel. The proposed action contains three separate management alternatives for adopting the policy. These management alternatives each contain different implementation alternatives related to that specific management alternative. The second policy alternative is the No Action Alternative which would involve no action by the United States in relation to the foreign research reactor spent nuclear fuel.

Each management alternative would result in very different policy considerations. Much of the foreign research reactor spent nuclear fuel analyzed in this EIS contains highly-enriched uranium (HEU), which can be used to make nuclear weapons. By adopting a policy to manage the foreign research reactor spent nuclear fuel, the proposed action would promote the U.S. goal of nuclear weapons nonproliferation by removing large amounts of HEU from civilian commerce. The No Action Alternative would be in direct conflict with the stated U.S. nuclear weapons nonproliferation goal and would seriously undermine credibility of the United States as a reliable partner in international nuclear weapons nonproliferation activities. Further, foreign research reactor operators may accuse the United States of failing to comply with its obligations under Article IV of the Non-Proliferation Treaty to share the benefits of peaceful nuclear cooperation with other countries.

Each management alternative would also result in very different environmental impacts in the United States which may vary according to the implementation alternatives of each management alternative. The No Action Alternative would have no direct environmental impacts in the United States.

Each of the three management alternatives under the proposed action is briefly summarized here. The three management alternatives were described in greater detail in Chapter 2, Sections 2.2 through 2.4. The policy considerations and environmental impacts of each alternative are described in detail in this chapter.

Management Alternative 1 — Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

Management Alternative 1 of the proposed action entails acceptance and management of the foreign research reactor spent nuclear fuel in the United States. This management alternative would have direct environmental impacts in the United States.

Management Alternative 1 is composed of nine basic implementation components, as well as seven implementation alternatives that alter one of these basic components in some manner. The basic implementation of Management Alternative 1, as well as the seven implementation alternatives, are described in detail in Chapter 2, Section 2.2. The policy considerations and environmental impacts of the basic implementation of Management Alternative 1 are presented in Section 4.2. The policy considerations and environmental impacts of the seven implementation alternatives of Management Alternative 1 are presented in Section 4.3.

Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Management Alternative 2 of the proposed action entails U.S. facilitation of overseas management of the foreign research reactor spent nuclear fuel at one or more foreign locations. No foreign research reactor spent nuclear fuel would be accepted into the United States. This would require advance negotiations and agreements with foreign reactor operators, officials in foreign governments, and reprocessing facilities. The outcome of these negotiations is uncertain. This management alternative would have no direct environmental impacts in the United States, unless the Department of Energy (DOE) decides to accept vitrified high-level waste from reprocessing facilities overseas in place of the foreign research reactor spent nuclear fuel. Very few countries have the capability to accept and store high-level wastes (GAO, 1994).

Management Alternative 2 is described in detail in Chapter 2, Section 2.3. Under this management alternative, the United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the foreign research reactor spent nuclear fuel containing U.S.-origin HEU. The policy considerations and environmental impacts of the two subalternatives of Management Alternative 2 are presented in Section 4.4.

Management Alternative 3 — Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

Management Alternative 3 entails some combination of the elements from Management Alternatives 1 and 2, and is referred to as the Hybrid Alternative. Management Alternative 3 would likely have more direct environmental impacts in the United States than Management Alternative 2, but less than Management Alternative 1.

Management Alternative 3 is described in detail in Chapter 2, Section 2.4. For purposes of analysis, a sample Hybrid Alternative has been included to demonstrate one possible combination of elements within Management Alternatives 1 and 2, and to allow an analysis of its impacts. It is important to note that the Hybrid Alternative described is merely an example for analysis purposes, and is only one of numerous possible combinations of elements from Management Alternatives 1 and 2.

Under the Hybrid Alternative described, DOE and the Department of State would facilitate the reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay, United Kingdom or Marcoule, France) for foreign research reactor operators in countries that can accept the reprocessing waste, as in Management Alternative 2. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1. The policy considerations and environmental impacts of the sample Hybrid Alternative (Management Alternative 3) are described in Section 4.5.

Other Alternatives and Comparisons

The No Action and Preferred Alternatives are discussed in Sections 4.6 and 4.7, respectively. Comparisons across all the alternatives of the potential impacts and costs are presented in Section 4.8 and 4.9, respectively. Finally, this chapter concludes by comparing the risks due to the alternatives to the risks due to other common activities in Section 4.10.

4.1 Overview of Environmental Impacts

4.1.1 Presentation of the Environmental Impacts

Potential environmental impacts associated with each segment of the affected environment of the proposed action are addressed in this chapter. These segments are presented in this section in the following order:

- Marine transport impacts,
- Port of entry impacts,
- Ground transport impacts, and
- Management Site impacts.

The impact analyses of these four segments are described in more detail in Appendices C, D, E, and F, respectively. Effects of each implementation alternative of Management Alternative 1 of the proposed action on U.S. nuclear weapons nonproliferation goals and objectives are also discussed. In addition, this chapter summarizes the potential costs associated with the alternatives. Details on costs are presented in Appendix F.

Spent nuclear fuel is transported in strong, heavy casks (NRC, 1993). After the spent nuclear fuel is delivered, the empty casks must be transported back on a return trip. Under most of the alternatives, empty casks would be transported overland, through U.S. ports, and on ships. There would be minor nonradiological impacts (vehicle emissions and potential traffic accidents) during ground transport of empty casks. These nonradiological ground transport impacts are included as part of the assessment in this EIS.

4.1.2 Key Assessment Factors

A key assessment factor is one that may differentiate among alternatives, has a measurable impact, or be of public interest. The detailed analysis of potential environmental impacts presented in the appendices of this EIS did not reveal any factor likely to cause a large impact. Because radiation exposure and its consequences is a topic of great public interest, emphasis is placed upon exposure to radiation, although DOE considers the evaluated effects of radiation to be small.

During handling operations, the principal hazard would come from radiation being emitted by the foreign research reactor spent nuclear fuel. Without adequate shielding, the radiation levels at the surface of some of the spent nuclear fuel itself would often be high enough to induce a prompt fatality. This radiation can and would be attenuated (i.e., reduced) by the shielding materials of the transportation cask, such as lead, steel, and polyethylene. Further, since radiation intensity decreases with distance, maintaining a distance from the cask would also provide radiation protection. At 100 m (330 ft) from the cask, the radiation levels would not be detectable above background radiation. All foreign research reactor spent nuclear fuel handling at the proposed foreign research reactor spent nuclear fuel management sites would take place at considerable distances from the public (greater than 100 m or 330 ft). Recently, actual radiation measurements were taken by the State of North Carolina, Department of Environment, Health, and Natural Resources, of the casks used in the first shipment of the 153 spent fuel elements covered by the Urgent Relief Environmental Assessment (DOE, 1994m). In every case, the State of North Carolina reported detecting no radiation above background levels (radiation exposure from natural sources) at a distance of 1 meter (3.3 ft) from the package surface (State of North Carolina, 1994).

Accidents involving foreign research reactor spent nuclear fuel could potentially also result in releases of radioactive material which could cause radiation exposures. For most accidents, essentially none of the radioactive material would be released because it is an integral part of the solid fuel. Larger quantities of radioactive elements could be released only when the accident generates enough energy to release particles of foreign research reactor spent nuclear fuel to the atmosphere, such as with a fire. However, the probability of such accidents is very small. For most accidents, the energy would not be high enough to damage the foreign research reactor spent nuclear fuel, so that none of the radioactive material would be released.

4.1.3 General Radiological Health Effects

The effect of radiation on people depends upon the kind of radiation exposure (alpha and beta particles, and gamma and x-rays) and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as absorbed dose. The sum of the absorbed dose to each tissue, when multiplied by certain quality and weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as effective dose equivalent (EDE).

An individual may be exposed to radiation from outside the body, or from inside the body because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The annual radiation dose limit to the public from nuclear facilities operated by DOE is 100 mrem per year (NRC, 1991). The potential foreign research reactor spent nuclear fuel management sites covered by DOE operations normally operate such that the public's dose is undetectable. For comparison, it is estimated that the average individual in the United States receives a dose of about 350 mrem per year from all sources combined, including natural and medical sources of radiation and radon. A modern chest x-ray, for example, results in an approximate dose of 8 mrem, while a diagnostic hip x-ray results in an approximate dose of 83 mrem (DOE, 1995c).

Radiation can also cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect for depicting the consequences of environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of cancers that may lead to death in later years. This effect is referred to as latent cancer fatalities (LCF) because the cancer may take years to develop and for death to occur, and may never actually be the cause of death.

In addition to LCF, other health effects could result from environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Table 4-1 shows the dose-to-effect factors for these potential effects as well as for LCF. For simplicity, this EIS presents estimated effects of radiation only in terms of LCF. The nonfatal cancers and genetic effects are less probable consequences of radiation exposure, and are less serious.

Table 4-1 Risk of LCF and Other Health Effects from Exposure to Radiation

Population[a]	*LCF*[b]	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Workers	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] *The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in sensitive age groups (that is, less than 18 years of age and more than 65 years of age).*

[b] *When applied to an individual, units are lifetime probability of LCF per rem of radiation dose. When applied to a population of individuals, units are excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.*

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. This is referred to as a "population dose." The total population dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1.0 person-rem (1,000 persons x 0.001 rem = 1.0 person-rem). The same population dose (1.0 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons x 0.002 rem = 1 person-rem).

The factor used in this EIS to relate a dose to its effect is 0.0004 LCF per person-rem for workers and 0.0005 LCF per person-rem for individuals among the general population (DOE, 1995c). The latter factor is slightly higher because of some individuals in the public, such as infants, who may be more sensitive to radiation than workers. These factors are based on the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP, 1991), and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (NRC, 1991). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to LCF are doubled. At much higher doses, prompt effects, rather than LCF, may be the primary concern. Unusual accident situations that may result in high radiation doses to individuals are considered special cases. No such cases are expected with either incident-free handling or accidents with foreign research reactor spent nuclear fuel.

These concepts may be applied to estimate the effects of exposing a population to radiation. For example, if 100,000 people were each exposed only to background radiation (0.3 rem per year), 15 LCF per year would be expected (100,000 persons x 0.3 rem per year x 0.0005 LCF per person-rem = 15 LCF per year).

Sometimes, calculations of the number of LCF associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1.0. For example, if 100,000 people were each exposed to a total dose of only 1 mrem (0.001 rem), the population dose would be 100 person-rem, and the corresponding estimated number of LCF would be 0.05 (100,000 persons x 0.001 rem x 0.0005 LCF per person-rem = 0.05 LCF).

The *average* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people is 0.05. In most groups, nobody (zero people) would incur an LCF from the one mrem dose each member would have received. In a small fraction of the groups, one latent fatal cancer would result; in exceptionally few groups, two or more latent fatal cancers would occur. The average number of deaths over all the groups would be 0.05 latent fatal cancers (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is zero LCF.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of LCF" corresponding to a single individual's exposure to 0.3 rem per year over a (presumed) 72-year lifetime is:

> 1 person x 0.3 rem per year x 72 years x 0.0005 LCF per person-rem = 0.011 LCF or one chance in 91 of an LCF.

Again, this should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual would incur a latent fatal cancer. Alternatively, this method estimates that about 1 person in 91 would die of cancers induced by background radiation.

4.1.4 Risks

Another concept important to the presentation of results in this EIS is the concept of risk. Risks are most important when presenting accident analysis results. The chance that an accident might occur during the conduct of an operation is called the probability of occurrence. An event that is certain to occur has a probability of 1.0 (as in 100 percent certainty). If an accident is expected to happen once every 50 years, the frequency of occurrence is 0.02 per year (1 occurrence every 50 years = 0.02 occurrences per year). A frequency estimate can be converted to a probability statement. If the frequency of an accident is 0.02 per year, the probability of the accident occurring in a 10-year program is 0.2 (10 years x 0.02 occurrences per year).

Once the frequency (occurrences per year) and the consequences (for radiation effects, measured in terms of the number of LCF caused by the radiation exposure) of an accident are known, the risk can be determined. The risk per year is the product of the annual frequency of occurrence times the number of LCF. This annual risk expresses the expected number of LCF per year, taking account of both the annual chance that an accident might occur and the estimated consequences if it does occur.

For example, if the frequency of an accident were 0.2 occurrences per year and the number of LCF resulting from the accident were 0.05, the risk would be 0.01 LCF per year (0.2 occurrences per year x 0.05 LCF per occurrence = 0.01 LCF per year). Another way to express this risk (0.01 LCF per year) is to note that if the operation subject to the accident continued for 100 years, one LCF would be likely to occur because of accidents during that period. This is equivalent to 1 chance in 100 that a single LCF would be caused by the accident source for each year of operation. This risk can be related to the risk of death from other accidental causes for comparison. As an example, the risk of dying from a motor vehicle accident is about 1 chance in 80. Similarly, the risk of death for the average American from fire is approximately 1 chance in 500, and for death from accidental poisoning, the risk is about 1 chance in 1,000 (NNPP, 1993). Section 4.10 compares the risks calculated in this EIS to those of common activities.

4.1.5 Estimated Radiation Dose Rate Near the Foreign Research Reactor Spent Nuclear Fuel Transportation Casks

The regulatory external radiation dose rate limit for foreign research reactor spent nuclear fuel transportation casks selected for use in the marine and ground transport analysis is 10 mrem per hour at 2 m (6.6 ft) from the "exclusive use" vehicle (no other cargo) [49 Code of Federal Regulations (CFR) 173.441]. This is equivalent to approximately 23 mrem per hour at 1 m (3.3 ft). Historical data from actual cask shipments of research reactor spent nuclear fuel have shown dose rates considerably below this regulatory limit. Dose measurements of casks containing research reactor spent nuclear fuel, including the foreign research reactor spent nuclear fuel recently received under the Urgent Relief Environmental

Assessment (DOE, 1994m), are presented in Appendix F, Section F.5. The average of these measurements is 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the cask. Recent measurements taken by the State of North Carolina on foreign research reactor spent nuclear fuel shipment packages, covered by the Urgent Relief Environmental Assessment, showed that the external dose rate at 1 m (3.3 ft) was undetectable above background radiation levels (State of North Carolina, 1994).

To be conservative, the analyses in this chapter use the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the side of the transport vehicle for the radiation dose rate near the foreign research reactor spent nuclear fuel casks. This conservative value was used in the calculations of incident-free doses to members of the public, marine transport workers, port workers, and ground transport workers. For radiation workers at the spent nuclear fuel management sites, the dose rate in the vicinity of the casks was estimated by the conservative methodology presented in Appendix F, Section F.5.

4.1.6 The Effects of Radiation on Plants and Animals

There is no convincing evidence from the scientific literature that chronic radiation doses below 1 rad per day will harm animal or plant populations. It is highly probable that limitation of the exposure of the most exposed humans (the critical human group, living on and receiving full sustenance from the local area) to 100 mrem per year will lead to dose rates to plants and animals in the same area of less than 1 rad per day. DOE and NRC regulations limit annual human exposures to values far lower than those that have caused observable damage in plant and animal populations. Therefore, specific radiation protection standards for nonhuman biota are not needed (IAEA, 1992).

4.2 Management Alternative 1 – Manage Foreign Research Reactor Spent Nuclear Fuel in the United States – Basic Implementation

This section presents the policy considerations and potential environmental impacts of the basic implementation of Management Alternative 1. Under the basic implementation of Management Alternative 1, all the foreign research reactor spent nuclear fuel could be accepted into the United States. DOE and the Department of State believe this would promote the nuclear weapons nonproliferation objective of reducing, and ultimately eliminating, civil commerce in HEU. The spent nuclear fuel could be managed safely and securely at any of five DOE sites.

Policy Considerations

A critical result of this basic implementation of Management Alternative 1 would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs. The successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States commitment to action. Finally, this basic implementation of Management Alternative 1 would support the Administration's nuclear weapons nonproliferation objective of not encouraging reprocessing for either nuclear power or nuclear explosive purposes.

Another crucial consideration associated with Management Alternative 1 is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other

Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation of Management Alternative 1 is up to approximately 19.2 metric tons of heavy metal (MTHM) representing approximately 22,700 elements. This amount is an upper limit because if some nations were to reprocess their research reactor spent nuclear fuel, for example, the amount of foreign research reactor spent nuclear fuel accepted into the United States would be reduced. Under the basic implementation of Management Alternative 1, approximately 4.6 metric tons (5.1 tons) of HEU would be removed from international commerce.

4.2.1 Marine Transport Impacts

Because the basic implementation of Management Alternative 1 involves ocean transport, DOE and the Department of State considered the environmental impacts on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) in accordance with Executive Order 12114 (U.S. Federal Register, 1979).

4.2.1.1 General Assumptions and Analytic Approach

The basic implementation of Management Alternative 1 includes the shipment of approximately 837 transportation casks containing foreign research reactor spent nuclear fuel over a 13-year period. Of these, approximately 721 transportation casks would be transported by sea to the United States, with the remainder (116) coming overland from Canada. DOE would prefer to consolidate the approximately 721 casks on board ships to minimize the number of voyages, but it is also possible that approximately 721 voyages could be required. This section evaluates the impacts of the marine transportation, including shipment in international waters from the port of origin to the United States and coastal shipping in United States territorial waters.

Four types of commercial cargo ships are considered to be candidates to carry foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1: containerized, breakbulk (general cargo), roll-on/roll-off, and purpose-built vessels (see Appendix C for a more complete description of these vessels). DOE and the Department of State assumed that all casks would be transported in standard International Standards Organization 20-ft shipping containers, because this is current shipping practice.

Nonradiological impacts associated with the marine shipment of 721 containerized transportation casks would be minimal. The United States receives more than 56,000 ships engaged in foreign trade at its ports each year (DOC, 1994). Shipping an additional 56 containers per year on average over the 13-year receipt period is not likely to cause any additional ships to sail beyond the number already scheduled. In the event that chartered vessels are used for this program, up to 10 voyages per year could be required, which is only 0.02 percent of the number engaged in regular commerce. Additional nonradiological impacts would be very small whether chartered or regularly scheduled commercial vessels are used. The number of containers handled on a regular basis is so large that the addition of the foreign research reactor spent nuclear fuel containers would add essentially no impacts (cargo vessels typically carry 800 to 1,000 containers per voyage). While nonradiological marine events such as unloading or cargo shifting accidents would be possible, the nonradiological impacts would be miniscule.

The radiological impacts of transporting the foreign research reactor spent nuclear fuel by sea were considered in two ways, incident-free impacts and accident impacts. The incident-free impacts would be those that occur simply due to the marine shipping of foreign research reactor spent nuclear fuel, assuming there are no accidents. The ship's crew would be the affected individuals in this case. The accident impacts would be the consequences of reasonably foreseeable accidents that might occur. These two evaluations are discussed in the following two sections, with additional details in Appendix C.

4.2.1.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Marine Transport

The primary impact of incident-free marine shipping of foreign research reactor spent nuclear fuel would be upon the crews of the ships used to carry the spent nuclear fuel casks. Since the crew of a ship is normally separated from the cargo and shielded by both the cargo and the ship's structure, the risk to the crew from spent nuclear fuel transport during most crew activities would be extremely low (DOE, 1994m). The exceptions would include the exposure to the crew during loading and off-loading of the spent nuclear fuel ISO containers and during daily inspection of the ship's cargo, including the containers housing the spent nuclear fuel transportation casks. Therefore, the crew exposure during loading, daily inspection, and unloading of the transportation casks has been incorporated into the incident-free marine transport analysis. The exposure to dock workers at the foreign research reactor spent nuclear fuel port of entry is assessed in Section 4.2.2.

Daily inspections of the casks is the activity that would result in the largest doses to the ship's crew, with the inspectors considered the maximally exposed workers during incident-free marine transport. For any given voyage, DOE and the Department of State conservatively assumed that the same three inspectors would conduct all of the inspections. The impact on the inspectors would be a function of the number of inspections performed, which would depend upon the amount of time the cask is onboard. Therefore, the incident-free radiological impact on the inspectors would depend upon the total duration of the voyage, including days at sea, in intermediate ports, and days in coastal sailing between intermediate ports. The duration of the voyage was selected as the weighted average of the duration of all the shipments necessary for 721 transportation casks. (See Appendix C for further details regarding this assumption.)

To maximize the estimated impact from incident-free transport, DOE and the Department of State made conservative assumptions regarding crew exposure. Specifically, DOE and the Department of State conservatively assumed that eight and two casks (loaded two casks per hold) would be shipped per voyage of chartered and regularly scheduled commercial ships, respectively. This assumption would result in additional exposure of the ship's crew due to the effect of loading casks into holds where a loaded cask would have already been stowed, and would also increase the exposure to the crew members performing daily inspections. The additional exposure would be a result of the combination of the radiation fields surrounding each of the transportation casks.

Assuming 56 casks per year, the number of annual voyages required would range from 7 to 28, depending upon the number of casks per ship. Although the foreign research reactor spent nuclear fuel would be shipped from 40 countries worldwide and to both U.S. coasts over a 13-year receipt period, DOE and the Department of State conservatively assumed that a single crew could be involved in up to 9 voyages per year. As a practical matter, this overstates the rate at which a crew would sail from Europe or Asia and back. Additionally, to determine the dose to the maximally exposed worker in the ship's crew, DOE and the Department of State conservatively assumed that the same individuals would conduct all the daily onboard inspections.

The dose received during daily cargo inspection would be a major contributor to the crew dose, so the duration of the voyage is an important consideration. Chartered vessels would sail directly to the port(s) of entry, yielding an average voyage duration of 18 days. DOE and the Department of State conservatively assumed that all shipments aboard regularly scheduled commercial breakbulk vessels would include two intermediate port stops in the United States, which would add 3 days to the voyage.

Table 4-2 presents the maximum estimated incident-free marine transport doses and risks. Values are provided for a chartered ship (which would not make intermediate port calls) and for a regularly scheduled commercial vessel. The values are based on the estimated time the cask would be onboard multiplied by the dose per day received as a result of inspections, plus the crew dose due to the foreign research reactor spent nuclear fuel container loading and off-loading activities. While the use of a chartered ship would result in higher per-shipment impacts (eight casks per shipment versus two for regularly scheduled commercial ships), the reduced number of voyages would offset this increase in per-shipment impacts. Therefore, the use of chartered ships instead of regularly scheduled commercial ships would result in slightly lower total crew exposures in the basic implementation of Management Alternative 1. The selection of the shipping mode, however, would not be based on crew exposures alone. Other factors, such as cost, would also be important in the choice of chartered or regularly scheduled ships. The results in Table 4-2, therefore, provide an estimate of the range of maximum worker exposures due to the shipment of the foreign research reactor spent nuclear fuel.

Table 4-2 Incident-Free Marine Transport Impacts[a]

	Regularly Scheduled Commercial Ship				*Chartered Ship*			
	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Risk (LCF)*	*Maximally Exposed Worker Dose (mrem)*	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Risk (LCF)*
Impacts Per Shipment	66[b]	0.000027	0.23	0.000091	100[b]	0.00004	0.83	0.00033
Impacts for the Basic Implementation	1,300[b,c]	0.00052[c]	85	0.034	1,300[b,c]	0.00052[c]	75	0.030

[a] *These results are based on the assumption that the dose rates associated with the casks are all derived from the exclusive-use regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of this regulatory limit, so this assumption is conservative.*

[b] *If an individual works on repeated shipments, this maximally exposed worker dose could exceed the annual regulatory limit. Therefore, DOE would require that mitigation measures be implemented to keep the maximally exposed worker dose down to 100 mrem per year or lower. See Appendix C for estimates of the total exposure to the ships' crews without mitigation measures.*

[c] *These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years.*

Marine transport workers are not trained to be radiation workers, so they would not be subject to the radiation worker limit of 5,000 mrem/yr. The applicable regulatory limit for these workers would be the same as for the general public: 100 mrem/yr. As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest estimated population risk is about 0.034 LCF, which is much less than 1 LCF.

4.2.1.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Marine Transport

The basic implementation of Management Alternative 1 could potentially impact the marine environment in the event of an accident involving the release of radioactive material from the spent nuclear fuel. This section discusses possible accidents and their consequences.

The range of accidents that could occur during marine transport is quite broad. The ship could collide with another vessel or an object such as a shoal, rock, or wreck. Foul weather could damage or sink the ship, or the ship could experience a fire, explosion, or other problem. To reduce the risk due to potential accidents, the casks that would carry the foreign research reactor spent nuclear fuel have been designed to prevent damage to the cask contents in all but the most severe, and least likely, cases. See Appendix B of this EIS for a description of the foreign research reactor spent nuclear fuel transportation casks.

Two scenarios emerge that could potentially threaten the marine environment and possibly humans: the cask could be damaged and then involved in a fire, or the cask could sink. These cases are discussed in more detail below.

Cask Damaged Followed by a Fire

A ship carrying foreign research reactor spent nuclear fuel could be involved in a severe collision with another ship. It is possible that a transportation cask, carried on a ship involved in such a collision, could be exposed to impact forces resulting from the collision. In that event, the cask could be damaged. However, only a small fraction, at most, of the force generated in a collision of one ship with another would be brought to bear on a transportation cask for two reasons. First, the force of a ship-to-ship collision would be distributed over the entire area of contact between the two ships, which means that the force density (force per unit of area) that would result from a collision must be considered. The maximum cross sectional area presented by a transportation cask would be small in comparison to the typical impacted area, so that even if a cask were located directly in the path of the collision and unprotected by intervening hulls, bulkheads, etc., the force that might be exerted on such a cask would be limited by the force density.

Secondly, ships floating on water are yielding objects, so that some portion of the energy of impact would be transmitted to the water. Even severe collisions with large impact forces, by themselves, would not necessarily result in catastrophic failure of a transportation cask. Thus, it would be even more unlikely for a less severe collision to result in the breach of a cask and, thereafter, a release of any of its contents. Attachment 4 to Appendix D discusses in detail the forces involved in ship collisions.

If it is assumed, however, that a ship collision breaches the cask, a release of radioactive material would be possible. In such a circumstance, the release would be small because the spent fuel is metallic and thus would release very limited quantities of radioactive material, even if mechanically damaged. However, due to the severity of the collision required to breach the cask, the ship carrying the foreign research reactor spent nuclear fuel cask would be severely damaged and probably would sink. Whether the ship would sink or not, the only humans that could be affected (the crew) would most likely not be in the vicinity of the impact point of the collision, where the damaged cask would be located.

The limiting accident is a ship collision severe enough to breach a cask carrying foreign research reactor spent nuclear fuel and also cause a large fire. Some of the radioactive contents of the cask could be released and carried into the air by the heated gases of the fire as a plume of radioactive particles. For an airborne release of this type to occur, the cask-carrying vessel must stay afloat during and immediately after the accident. In practice, this would mean that the ship must stay afloat for a period of some hours following an accident of the requisite severity. This latter condition must be satisfied for atmospheric dispersal to occur, even though marine casualty files indicate that a common outcome of severe ship collisions is rapid sinking, often within a matter of minutes. Assuming the cask was damaged by a severe collision; and the ship remained afloat despite the severe collision; and the cask was engulfed in flames for a time sufficient to release a radioactive plume, there would likely be no human population on the ocean (excluding the crew) who could be affected.

It is possible that the ship could be in coastal waters (i.e., beyond the port's sea buoy) at the time of this severe collision. Except in port, a ship is seldom within 16 km (10 mi) of a population center, so the port accident public risk analysis in the next section covers public risk in this scenario. The ship's crew and people onboard other vessels that may come to provide assistance could be exposed to any released radioactive material. The number of people potentially exposed would be less than that used in the port accident analysis for populations near a port [less than 1.6 km (1 mi) from the port]. Additionally, accident frequencies at sea tend to be lower than in-port accident frequencies. Therefore, both the consequences and risks for an accident at sea are covered by the results of the port accident analysis.

Risks associated with this type of accident at sea are covered by the risks of the same type of accident in ports because humans in the vicinity of the accident at sea are much fewer in number than even the least populated port.

Sunken Cask

The second scenario of concern is that a foreign research reactor spent nuclear fuel cask or casks would be sunk. This could be the result of the ship sinking, of the casks being somehow swept overboard, or of a ground transport accident on a causeway. Submersion of an intact cask would not necessarily result in a release of its contents, as spent nuclear fuel casks are designed to withstand at least a 15 m (50 ft) immersion. It has been demonstrated that cask seals will remain intact at much greater depths (DOE, 1994m). Should a loaded foreign research reactor spent nuclear fuel cask (damaged or undamaged) sink anywhere in the U.S. coastal waters, it will be recovered regardless of depth. U.S. Coastal waters in this case refers to waters within the 12 mile territorial limit. Recovery would be accomplished, even in the deepest parts of U.S. coastal waters, such as in Puget Sound, which reaches 305 meters or 1,000 feet (Encyclopedia Americana, 1991). Elsewhere in the world, spent nuclear fuel casks can, and likely would, be recovered from water up to 200 m (660 ft) deep, which is beyond the range typical of coastal and port depths. Typically 200 m (660 ft) is considered the limit of the continental shelf. Recovery at depths greater than 200 m (660 ft) is possible but is more difficult.

If a sunken cask containing foreign research reactor spent nuclear fuel were recovered, the effect on the marine environment would be minimal, even if the recovery effort required up to 1 year to complete. The release to the ocean water of radioactive particles from the spent nuclear fuel requires that first the metallic spent fuel corrode, then the radioactive particles escape from the cask. Even if the cask were damaged, the most likely damage to a spent nuclear fuel cask, either from mechanical trauma or excessive depth, would be failure of the seal. Seal failure would allow seawater to enter the cask to begin the corrosion of the metallic spent nuclear fuel, but the flow of water through the cask to carry out the radioactive material would be minimal due to the small cross sectional area of the failed seal. The decay heat from the spent nuclear fuel is low, thereby providing no driving force to expel water out of the cask through the failed seal.

If a cask was not recovered, the radioactive constituents of spent nuclear fuel would be released slowly over time into the surrounding waters. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a cask were submerged on the deep ocean bottom and not recovered, the peak human dose to an individual ingesting seafood harvested from the area in which the breached submerged spent nuclear fuel cask would be located would be 114 mrem per year. If a sunken cask in coastal waters was not recovered, the peak human dose is conservatively estimated to be 14,000 mrem per year. Consequences to humans and to marine biota are presented in Table 4-3. Other studies of similar circumstances indicate that the individual dose would be even lower (DOE, 1980). Uranium (the major constituent of the spent nuclear fuel) has been found not to bioaccumulate in fish, and bioaccumulates only slightly in crustaceans and mollusks (IAEA, 1976). The peak doses for humans, fish, crustaceans, and mollusks are presented in Table 4-3 in the situation where a chartered ship carrying eight casks might sink in deep ocean. Doses for humans and other animals are expressed in units of rem and rad, respectively. Rem is discussed in some detail in Section 4.1.3. While rem is only used for measuring human exposure to radiation, rad is used to measure exposure of nonhumans to radiation. Rad is a unit of absorbed dose from ionizing radiation.

The probability provided in Table 4-3 is the probability of one ship accident and loss of a cask during the entire program. The consequences are from one unrecovered cask. The program risk is the product of the probability and the consequences. Humans would not be the principally exposed species in a marine accident involving foreign research reactor spent nuclear fuel. Estimates were made of the dose to the biota received from a damaged cask containing foreign research reactor spent nuclear fuel. This analysis assumes that the cask would lay on the deep ocean floor where it would slowly release its radioactive inventory whether it was damaged in the collision or not.

Table 4-3 Impacts of Unrecovered Casks in Deep Ocean

	Probability	*Consequences*	*Program Risk*
MEI (human)	1.7×10^{-6}	114 mrem/yr	0.00019 mrem/yr
Fish	1.7×10^{-6}	640 rad/yr	1.1 mrad/yr
Crustaceans	1.7×10^{-6}	880 rad/yr	1.4 mrad/yr
Mollusks	1.7×10^{-6}	30,000 rad/yr	49 mrad/yr

Risks associated with the release of the contents of the spent nuclear fuel elements into the deep ocean are expected to be very small due to the low probabilities and limited consequences. The highest estimated risk to the MEI is 0.00019 mrem per year for every year that the cask leaks and this hypothetical individual ingests seafood harvested from near the cask. DOE and the Department of State assume that these conditions could apply for about 5 years, so the total MEI dose would be 0.00095 mrem. This translates into a maximum estimated MEI risk of 5×10^{-10} LCF. This means that this hypothetical individual's additional chance of incurring an LCF would be less than one in a billion. The risks to fish, crustaceans, and mollusks are low enough that no adverse impacts would be expected.

Probabilities, consequences, and risks were also calculated for the cases of unrecovered casks in coastal waters, both undamaged and damaged. The results are presented in Table 4-4, again in terms of rem for humans and rad for other animals. In coastal waters, cask recovery is considered likely (NEA, 1988), which makes the probabilities in Table 4-4 low. Comparing Tables 4-3 and 4-4 shows that the consequences of a sunken cask in coastal waters would be greater than in the deep ocean, but when multiplied by the probabilities, the risks are actually lower.

Table 4-4 Impacts of Unrecovered Casks in Coastal Waters

	Probability One Undamaged Cask	*Consequences One Undamaged Cask*	*Program Risk*
MEI (human)	2.3×10^{-8}	190 mrem/yr	4.3×10^{-6} mrem/yr
Fish	2.3×10^{-8}	77 mrad/yr	1.8×10^{-6} mrad/yr
Crustaceans	2.3×10^{-8}	81 mrad/yr	1.9×10^{-6} mrad/yr
Mollusks	2.3×10^{-8}	210 mrad/yr	4.8×10^{-6} mrad/yr
	Probability One Damaged Cask	*Consequences One Damaged Cask*	*Program Risk*
MEI (human)	4.6×10^{-11}	14,000 mrem/yr	6.4×10^{-7} mrem/yr
Fish	4.6×10^{-11}	620 mrad/yr	2.9×10^{-8} mrad/yr
Crustaceans	4.6×10^{-11}	660 mrad/yr	3.0×10^{-8} mrad/yr
Mollusks	4.6×10^{-11}	14,000 mrad/yr	6.4×10^{-7} mrad/yr

These risk estimates were derived assuming that the foreign research reactor spent nuclear fuel is shipped at a rate of one cask per voyage. Assuming a different shipping schedule, such as eight casks per voyage, would not result in a different estimate of the risks. The potentially higher consequences of an accident involving more than one shipping cask would be balanced by the reduced probability of an accident due to the reduced number of shipments. For example, the risk associated with one shipment of eight casks is equivalent to the risks associated with eight single cask shipments.

4.2.1.4 Marine Transport Cumulative Impacts

The cumulative impact of radioactive material shipments on ships' crews beyond that discussed in Section 4.2.1.2 was not estimated. In estimating the cumulative impact on port workers (see the following section) it was possible to estimate the total number of shipments of radioactive material through a port. However, it is not as simple to estimate the total number of shipments of radioactive material that involve the same ship and crew. It is expected that each ship's crew would be exposed to fewer of the shipments of radioactive material than that assumed for the port worker in the cumulative impact analysis for the port. For port workers, the impacts of the shipments other than the foreign research reactor spent nuclear fuel were of the same order of magnitude, but lower than the foreign research reactor spent nuclear fuel shipments. Therefore, the individual crew member's exposure from shipments other than the foreign research reactor spent nuclear fuel shipments would be a small fraction of the dose received due to the foreign research reactor spent nuclear fuel shipments.

4.2.1.5 Marine Transport Mitigation Measures

The principal environmental impact that would occur during marine transport would be radiation dose to the ships' crews. Most of this dose occurs because crew members must visually inspect the cargo every day for safety reasons, and the inspections cannot be curtailed.

The magnitude of the estimated impacts from this portion of the basic implementation of Management Alternative 1 is primarily due to two items: the conservative assumption that the radiation field emanating from all of the casks would be at the regulatory limit (as opposed to the levels of one-tenth of the regulatory limit that have been observed in past foreign research reactor spent nuclear fuel shipments), and the conservative assumption that the same crew member is involved in inspections for all of the casks on nine shipments during any given year. In reality, neither of these conservative assumptions would be likely to occur. Nevertheless, to ensure that no member of a ship's crew could receive a dose above what is allowed for a member of the general public, DOE would mitigate this effect by implementing a system through its shipping contractor to track each ship and crew involved in the shipment of foreign research reactor spent nuclear fuel. DOE would also include a clause in the contract for shipment of the foreign research reactor spent nuclear fuel requiring that other crew members be used if any crew member approaches a 100 mrem dose in any year.

If a cask or casks were sunk in deep ocean or coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2 Port Activities Impacts

4.2.2.1 General Assumptions and Analytic Approach

To assess the range of potential impacts on ports at which a ship carrying foreign research reactor spent nuclear fuel might call, 13 ports of entry representing a wide range of port city population densities were selected for detailed evaluation. Eight of the ports—Charleston, SC; Elizabeth, NJ (for the New York City area); Philadelphia, PA; Norfolk, VA (representing Hampton Roads); Jacksonville, FL; Savannah, GA; Wilmington, NC; and Military Ocean Terminal at Sunny Point (MOTSU), NC—are East Coast ports that represent high, medium, and low population density ports. The Norfolk Terminal was selected to represent the three terminals (Newport News, Norfolk, and Portsmouth) at Hampton Roads for the analysis of potential impacts because this terminal provides the most conservative values in terms of estimated impacts. The West Coast ports chosen for evaluation were Long Beach, CA; Concord Naval Weapons Station (NWS), CA; Portland, OR; and Tacoma, WA, to represent high and medium population density ports. On the Gulf Coast, Galveston, TX was analyzed. These ports were selected to represent a range of ports in this analysis, not necessarily as the chosen ports of entry for foreign research reactor spent nuclear fuel. Ports representative of a group of ports with similar characteristics (i.e., of similar population around the port) were selected for analysis rather than attempting to analyze accidents at every potential port. Actual port selection and specific selection criteria are discussed in Appendix D, Section D.1.

The analysis assumed that there were no restrictions on the shipping routes taken by the cargo vessel carrying the foreign research reactor spent nuclear fuel. This assumption allows the vessel to make intermediate stops at any U.S. port capable of unloading the vessel. This implies that the vessel could enter most ports capable of receiving ocean-going cargo vessels, a group of ports that far outnumbers the ports that survive the port selection criteria for the receipt of foreign research reactor spent nuclear fuel. It was conservatively assumed that regularly scheduled commercial ships carrying foreign research reactor spent nuclear fuel would pass through two intermediate U.S. ports before reaching the port of entry for the foreign research reactor spent nuclear fuel. The 13 ports with high, medium, and low population densities that were chosen for site-specific accident analysis provide a perspective on the accident risks at the more than 100 ports that could be intermediate ports of call for the foreign research reactor spent nuclear fuel vessels.

Each port stop would or could involve:

- Port entry from the sea buoy,
- Docking,
- Inspection of cargo,

- Partial unloading of cargo,
- Partial reloading of cargo, and
- Port exit to the sea buoy.

As with the marine transport, the port impacts were evaluated for two conditions: incident-free and accident conditions. Summary results are presented in the following sections. Details of the analysis are presented in Appendix D.

4.2.2.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Port Activities

As stated in Section 2.6, no spent nuclear fuel transportation cask has ever released its contents (radioactive material), even as a result of an accident. For this reason, release of radioactive material is not considered as part of the incident-free analysis. The only impact considered is that caused by radiation exposure due to radiation emitted by foreign research reactor spent nuclear fuel contained within the transportation casks. Since no radioactive material would be released, there would be no impacts on land, water, or air quality in any of the ports or any of the waterways used by ships in the transport of foreign research reactor spent nuclear fuel.

Risks associated with the foreign research reactor spent nuclear fuel in incident-free conditions in port are predominantly those to inspectors and port workers. Port workers and inspectors are not radiation workers as defined by NRC regulations. Thus, the maximum allowable annual exposure for these personnel would be 100 mrem, the same radiation dose limit established by the NRC to protect individual members of the public (DOE, 1990c). When a ship arrives in its first port, the spent nuclear fuel package would be inspected by customs officials, U.S. Coast Guard personnel, and others. Up to six inspections, estimated at up to 15 minutes per person per spent nuclear fuel cask, were conservatively assumed. Once inspections are complete, the ship would partially unload and reload cargo. After that, DOE and the Department of State conservatively assumed that the ship would proceed to another intermediate port and then to the port of entry for the foreign research reactor spent nuclear fuel.

To determine the incident-free risks associated with port operations, two types of ships were considered for the shipment of the foreign research reactor spent nuclear fuel. In the first case, DOE and the Department of State conservatively assumed that all shipments were made on regularly scheduled commercial breakbulk ships. This type of vessel was selected because it maximized the time required for port activities, such as off-loading and inspections. In addition, during the operations at the intermediate port stops, DOE and the Department of State conservatively assumed that other unloading and loading operations would occur in the vicinity of the container with the loaded foreign research reactor spent nuclear fuel cask in one of the intermediate ports. Risks associated with these activities, which are comparable to the risks associated with the off-loading of the foreign research reactor spent nuclear fuel, have been included in the assessment. Transport of the material on this type of vessel would therefore result in the highest worker radiation doses in the incident-free analysis. All worker exposures were calculated by estimating the times required for activities and the distances from the transportation cask to where the worker would most likely be located.

To provide a measure of the difference in the worker exposures resulting from the use of cargo vessels other than the regularly scheduled commercial breakbulk vessels, the analysis was also performed for port operations associated with the use of a chartered container vessel. This type of vessel requires the least amount of time to unload. DOE and the Department of State also assumed that a chartered vessel would not make any intermediate port stops, so that the ship's port of entry into the United States would also be the port of entry for the foreign research reactor spent nuclear fuel. Use of these two types of vessels in the analysis provides an estimate of the range of the maximum incident-free risk associated with port operations.

At the port of entry, the casks would be off-loaded by port workers, and arrangements would be made for the immediate departure of the foreign research reactor spent nuclear fuel from the port. In recognition of instances where some delay may occur, DOE and the Department of State conservatively assumed a delay of up to 24 hours in a secure staging area. The 24-hour period for the staging of spent nuclear fuel casks was selected because it is possible that, on occasion, the spent nuclear fuel casks would not leave the secure staging area the same day that they arrived, depending on variables such as the time of day the casks clear customs and the weather. Nonetheless, DOE and the Department of State consider it unlikely that the casks would remain in the staging area for longer than 24 hours.

To estimate the maximum individual exposure, the shipments were divided into East Coast and West Coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East and parts of South America were designated as East Coast shipments, all others were designated as West Coast shipments. Under these assumptions, the East Coast port(s) would receive approximately 535 casks and the West Coast port(s) approximately 186 casks. DOE and the Department of State also conservatively assumed for this analysis that all the shipments would pass through the same intermediate ports as the shipments on regularly scheduled commercial vessels and have the same port of entry.

Further, DOE and the Department of State made the very conservative assumption that the same inspectors and workers would handle every cask shipment. The per-shipment doses were then multiplied by the number of shipments for the East Coast to determine the maximally exposed worker dose for the basic implementation of Management Alternative 1.

In determining the worker population exposure, all shipments (East Coast and West Coast) were considered. This results in the integrated dose for the entire basic implementation of Management Alternative 1 which would span 13 years. The maximum estimated incident-free risks to port personnel due to the basic implementation of Management Alternative 1 are presented in Table 4-5. The incident-free risk to the general public would be zero because only workers would be near the casks in port.

This table shows the maximally exposed worker dose, worker population dose, and associated risks for the shipment of foreign research reactor spent nuclear fuel as containerized cargo on a regularly scheduled commercial breakbulk vessel and as cargo on a chartered container vessel. These figures represent the range of maximum estimated impacts for the various shipping modes available for the ocean transport of foreign research reactor spent nuclear fuel.

As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest total population risk for port workers is 0.012 LCF, which is much less than one LCF.

Table 4-5 Incident-Free Port Activity Impacts[a,b]

Impacts per Cask Transfer								
	Regularly Scheduled Commercial Breakbulk Ship				***Chartered Container Ship***			
Risk Group	***Maximally Exposed Worker Dose (mrem)***	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Workers per Cask (person-rem)***	***Population Risk (LCF)***	***Maximally Exposed Worker Dose (mrem)***	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Workers per Cask (person-rem)***	***Population Risk (LCF)***
Inspectors	3.8	0.0000015	0.013	0.0000052	1.3	5×10^{-7}	0.0053	0.0000021
Port Handlers, Intermediate Ports	2.2	9×10^{-7}	0.018	0.0000071	----	----	----	----
Port Handlers, Port of Destination	2.0	8×10^{-7}	0.0066	0.0000026	0.46	1.8×10^{-7}	0.0015	6×10^{-7}
Port Staging Personnel	0.36	1.4×10^{-7}	0.0045	0.0000018	0.4	2×10^{-7}	0.0046	0.0000018
Maximum	3.8	0.0000015	----	----	1.3	5×10^{-7}	----	----
Total	----	----	0.042	0.000017	----	----	0.011	0.0000045
Impacts for the Entire Basic Implementation								
	Regularly Scheduled Commercial Breakbulk Ship				***Chartered Container Ship***			
Risk Group	***Maximally Exposed Worker Dose (mrem)***	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Workers (person-rem)***	***Population Risk (LCF)***	***Maximally Exposed Worker Dose (mrem)***	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Workers (person-rem)***	***Population Risk (LCF)***
Inspectors	1,300[c]	0.00052[c]	9.4	0.0038	670	0.00027	3.8	0.0015
Port Handlers, Intermediate Ports	1,186	0.00047	13	0.0052	----	----	----	----
Port Handlers, Port of Destination	1,072	0.00043	4.8	0.0019	250	0.0001	1.1	0.00044
Port Staging Personnel	190	0.000076	3.2	0.0013	210	0.000084	3.3	0.0013
Maximum	1,300[c]	0.00052	----	----	670	0.00027	----	----
Total	----	----	30	0.012	----	----	8.2	0.0032

[a] *These results are based on the assumption that the dose rates associated with the casks are all based on the regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of the regulatory limit, so this assumption is conservative.*

[b] *These results are all based on the assumption that each voyage carries two casks. This assumption is conservative because chartered ships may carry up to eight casks.*

[c] *With all the conservative assumptions in this analysis, the maximally exposed worker dose could theoretically exceed the annual regulatory limit. Therefore, DOE would require mitigation measures to keep the maximally exposed worker dose down to 100 mrem per year or lower. These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. See Appendix D for maximally exposed worker doses without mitigation measures.*

4.2.2.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Port Activities

Section 4.2.1.3 discussed the impacts of marine accidents that could occur either in the open ocean or during coastal passages. This section discusses the impacts of accidents that could occur anywhere from the sea buoy into the port and at the pier.

Methodology

An analysis of reasonably foreseeable accidents must evaluate the consequences of possible accidents and the probability of an accident occurring. In incident-free marine transport, some exposure would be expected from radiation emitted from the casks. In the case of accidents, the probability of exposure is only an estimate of a hypothetical event. Accident probabilities are derived from published maritime accident rates. The analysis of ship collisions concludes that only one hold of the ship carrying the foreign research reactor spent nuclear fuel transportation casks would be subject to sufficient forces to potentially result in cask damage. There is no difference between the risks associated with a single shipment with two casks in a hold, and two shipments of a single cask each. The consequences of the accident with two casks in the hold may be as large as twice the consequences of an accident involving one cask. But the probability of an accident involving the ship carrying the two casks is half the probability of one of the two ships carrying a single cask being involved in an accident. Therefore, the potential risk from accidents, marine transportation of spent nuclear fuel was modeled in the port accident analysis as occurring in one cask per shipment.

Because accidents can be of any degree of severity, from a "fender bender" to one involving severe impact and prolonged fire, the severity spectrum is divided into a number of accident severity categories. Each category is assigned a conditional probability of occurrence [i.e., a probability (given that an accident occurs) that it will be of that particular severity]. In general, the more severe the accident, the more remote the chance of such an accident. In this analysis, the accident severity spectrum is divided into six categories (Wilmot et al., 1981), which are discussed further in Appendix D. The accident scenarios considered in this analysis fall into the three most severe of the six severity categories.

Accidents in the first three, least severe, categories result in no release of material from the spent nuclear fuel transportation cask. These categories include all the accident scenarios associated with handling the spent nuclear fuel cask, including dropping the cask during transfer from the vessel to the truck or train. The transportation casks are certified to maintain their integrity when dropped from 15 m (50 ft) onto a perfectly unyielding surface. During the cask transfer, however, the crane may lift the cask higher than 15 m (50 ft). As the dock surface is softer than "perfectly unyielding," the soft surface of the dock would compensate for the greater drop height. Studies (DOE, 1994m) have shown that a cask can be dropped from much higher than the certification test height onto a yielding surface, without breaching.

The accidents analyzed in the three highest severity categories include collision of vessels, either in the approach to the harbor or when the vessel transporting the foreign research reactor spent nuclear fuel would be docked. The category 4 accident severity category models a collision of two vessels resulting in the breach of the transportation cask. Severity categories 5 and 6 model collisions that would breach the cask and subsequent fires that would cause the release of additional material, with category 6 fires being more intense than those for category 5.

As mentioned above, the spectrum of accidents, including a container breach and fire, were evaluated at two locations in each of the 13 ports of entry selected to envelop the port impacts. The approach to each port, from the sea buoy to the selected dock, was examined to determine the location where the accident would be most likely as well as have the largest consequence. This point is typically near the highest population center along the approach to the pier, and DOE and the Department of State conservatively selected this point for accident analysis. The second location where the spectrum of accidents was assumed to occur is at pier-side.

At these two locations, the probability of an accident was assigned, based on historical ship accident data (see Appendix D for details). These data were used to develop accident frequencies for collisions between vessels large enough to generate the forces sufficient to damage the cask (additional details on the development of the model used are provided in Appendix D), and to develop the frequency of collisions concurrent with fires (Lloyds, 1991). These data include information on a large number of ship voyages and accidents due to all causes. The cause of the accident (human error, weather, mechanical failure, etc.) was not identified for this analysis. However, the data apply to damage to or loss of a vessel and would include information on accidents that were caused by severe weather. Although severe weather accident scenarios are not specifically identified in the analysis, they are considered through the use of these data.

The consequence modeling for the port accident analysis was performed using the MELCOR Accident Consequences Code System (MACCS) (Jow, 1990), a code developed for the conservative modeling of accident consequences for nuclear powerplants and approved by the NRC. This code uses site-specific information, including population and meteorology, along with the identified radionuclide inventory and release fractions to determine the consequences of the accident scenarios. In determining the effects of the release of radioactive material, the MACCS code evaluates the direct dose to the public as well as several additional pathways including inhalation, ingestion, and groundshine. Groundshine is the dose received from radioactive material deposited on the ground's surface.

A conservative assumption incorporated into the risk assessment is that the entire population would remain in the area for 24 hours and therefore would be exposed to the greatest extent possible to radioactive material deposited on the ground from the plume. In reality, individuals close to an accident could be evacuated.

Atmospheric dispersion is usually the primary mechanism for dispersing any material that might be released in a severe accident. For the ship-collision-without-fire scenario (category 4), the release is modeled as occurring at the water surface level. For shipboard fires (categories 5 and 6), an elevated release due to the lifting effect of the fire is modeled. Meteorology data from the nearest National Weather Service Station were obtained for the 13 ports of entry selected as representative ports and input to the analysis of dispersion to ensure validity.

Cask Characteristics

The behavior of the cask in accidents within each accident severity category is accounted for in this analysis. "Type B" spent nuclear fuel casks (the kind in which the foreign research reactor spent nuclear fuel would be shipped) are massive, highly damage-resistant packages. Moreover, the spent nuclear fuel itself consists mostly of solid metallic materials that are not readily dispersed. Therefore, large releases would not be likely to occur, even in the most severe of accident conditions.

"Type B" packages are required to pass a series of rigorous tests that are associated with hypothetical accident conditions that might be encountered. These certification tests were developed by the International Atomic Energy Agency and promulgated as model regulations (IAEA, 1990). These model regulations have been adopted by the United States as well as all of the nations currently proposing to ship foreign research reactor spent nuclear fuel to the United States under the basic implementation of Management Alternative 1.

Ports Selected for Accident Analyses

Analyses of the impacts of possible accidents at representative ports were conducted. Thirteen ports were selected as being representative of the full range of ports in the United States, based on population and geography. Three of the ports are high-population density ports, two on the East Coast (Elizabeth, NJ and Philadelphia, PA) and one on the West Coast (Long Beach, CA). Five of the ports (Portland, OR; Tacoma, WA; Concord NWS, CA; Jacksonville, FL; and Norfolk, VA) are medium-population density ports, three on the West Coast and two on the East Coast. The remaining ports (MOTSU, NC; Galveston, TX; Savannah, GA; Wilmington, NC; and Charleston, SC) are low-population density ports. The 13 potential ports of entry for which accidents were analyzed collectively have a range of populations and geography that ensure that the results of these analyses are representative of the results that would have been reached if the analyses had been conducted for all ports. Additionally, these 13 ports include all 10 of the ports that meet all of the port selection criteria.

To demonstrate the representative nature of the analyses performed, a plot was made of the analyzed accident consequences for mean meteorological conditions at each port versus the port's population in a 16-km (10-mi) radius (Figure 4-1). The analyzed data points are shown as dots. The straight line represents the linear least squares fit of the data. Since the straight line represents an average of the data, some deviation from the line for individual data points is expected. The data fit well, with a correlation factor of 0.994455 (a correlation factor of 1.0 implies a perfect fit). This plot demonstrates the expected increase in the total population dose with an increase in port population. Deviations from the line by the calculated data are typically due to the distribution of population in relation to the local meteorology. Where most of the population is downwind of the port in normal weather, the corresponding population dose would likely be above the average line. For comparison, the total population dose due to background radiation is shown in the upper right corner. This comparison shows that population dose resulting from a severe accident would be approximately 0.2 percent of the annual background population dose.

As a check that the data from the 16-km (10-mi) radius population is valid, a similar analysis was performed correlating the 80-km (50-mi) radius population and accident consequences for seven ports. This analysis confirmed that the population dose as a function of population is linear, and therefore confirms that the range of ports selected for accident analysis fully covers the entire range of U.S. ports. More specific discussion of the results of the analyses is provided in Appendix D. This linearity of consequences and population show that any port selected for use as an intermediate port or port of entry for the foreign research reactor spent nuclear fuel, ranging from the least populous port (MOTSU) to the most populous port (Elizabeth) and including major ports of intermediate population, has had representative accident analyses performed.

Probabilities of Port Accidents

The probability of an accident occurring can be determined from historical statistics on ship collisions and mishaps. Maritime accident rate data from a Lloyd's of London database covering approximately 900,000 port calls by commercial vessels over a 15-year period (1978 to 1993) were examined to develop accident probabilities. The data indicate that the basic accident rate in and near ports is slightly less than 0.0001 accidents per port transit, or approximately 1 accident per 10,000 port visits.

Only the most severe accidents, however, would cause any damage to the cask. Thus, the conditional probabilities of occurrence of each accident severity were also developed from this database. As discussed in Appendix D, a conditional probability is defined as the probability, given that an accident has occurred, that it will be of a certain severity. To calculate overall probability of an accident of a particular severity, the base accident probability (accident rate) must be multiplied by the conditional probability.

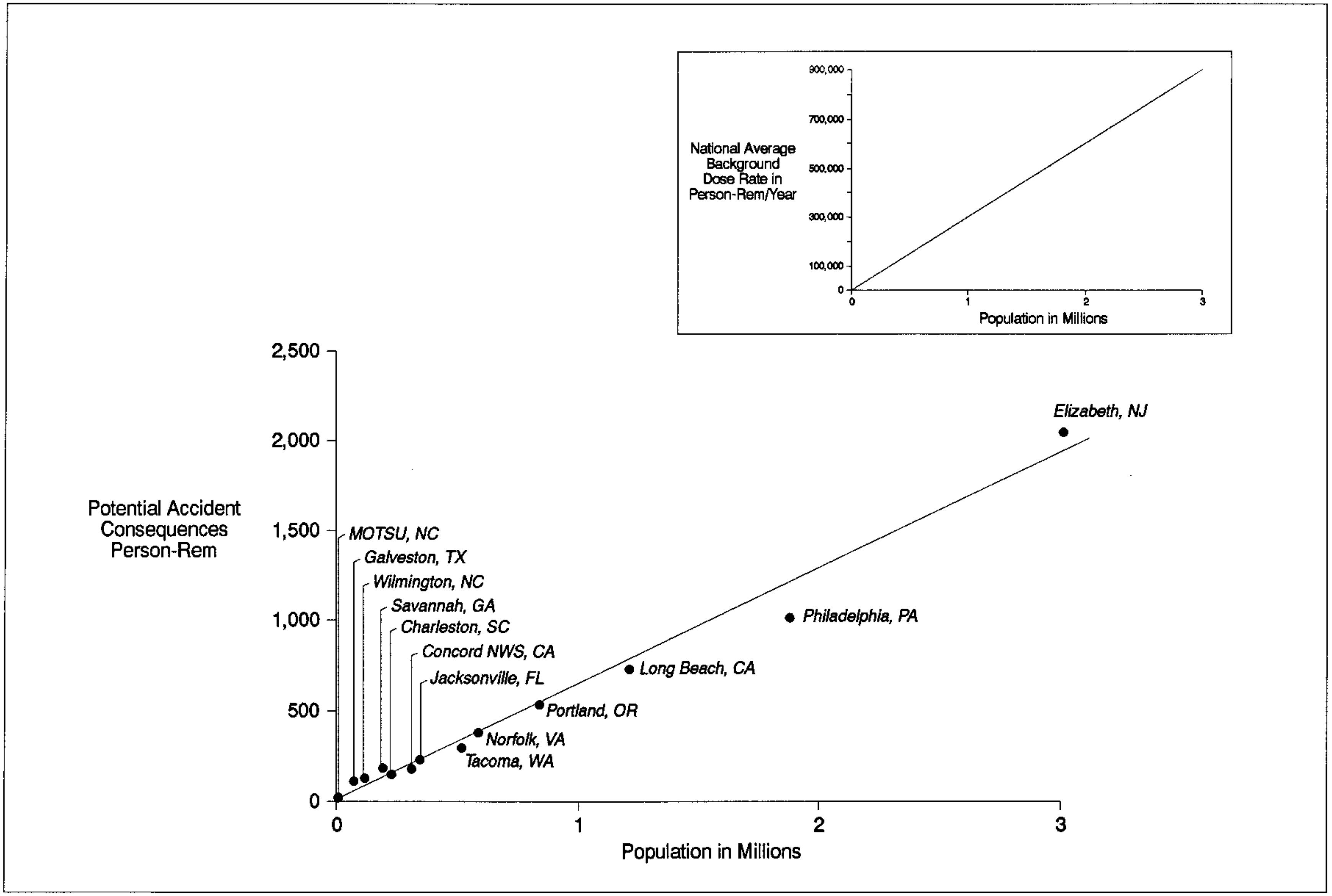


Figure 4-1 Consequences Versus Population [for a 16-km (10-mi) Radius]

Accidents are ranked according to their release categories. Release category 4 would result from the cask being damaged and compromised. Release category 5 would result from a damaged and compromised cask being enveloped in a fire. Release category 6 would result from a damaged and compromised cask being enveloped in a longer fire than a category 5 fire. The probabilities for the category 4, 5, and 6 accidents are 0.000006, 5×10^{-9}, and 6×10^{-10}, respectively. DOE and the Department of State assumed that it was equally likely that the accident occurs at the dock or in the channel, during the approach to the dock.

Consequences of Port Accidents

The consequence of an accident indicates the result, given that the accident were to occur, without any consideration for the likelihood of the accident occurring. The analysis conducted to determine the impacts of an accident involving foreign research reactor spent nuclear fuel in ports yields two different measures of the consequences. One measure is a calculation of the number of LCF that might result if the accident were to occur. These results are presented in Table 4-6 for the three most severe types of accidents under mean meteorological conditions.

The results presented in Table 4-6 are based on the mean consequences, so they are equivalent to results expected for the accidents in the respective release categories. These results are also based on the conservative assumption that accidents involve a cask carrying the highest inventory of nuclear material expected. Appendix D provides information on the consequences associated with the range of spent nuclear fuel types considered.

Examination of Table 4-6 shows that the most adverse consequence (2.9 LCF) arises from a Release category 5 accident in the channel approaching the Port of Elizabeth. This places the vessel just west (and generally upwind) of New York City. Although some of the release fractions change between categories 5 and 6, most of them do not. Therefore, the total population dose and the related number of LCF are about the same for Release categories 5 and 6. Release category 4 would be a release with no fire. In the absence of a fire, the release would remain at ground or water level without wide dispersion, hence the greatly reduced number of affected individuals and reduced consequences.

In addition to calculating the health effects of an accident on man, the MACCS code also calculates the impact of the accident on the land and structures around the accident site. These effects are characterized by the costs of activities required to bring the land and structures back into a usable condition. These activities are characterized as (1) no remedial action required; (2) decontamination – the resources can be returned to use immediately after clean-up; (3) interdiction – the resources must be temporarily abandoned, for several years, prior to their return to use; and 4) condemnation – the resources are considered unusable for an extended period. In all of the consequence analyses performed for each of the accident sites, there are no costs calculated that are associated with decontamination, interdiction, or condemnation. This means that all of the land and structures would be immediately available for use. (The consequences calculated by MACCS for the immediate vicinity of the accident are based on average value for the area within 1.6 km (1 mi) of the accident. Even though the average consequences calculated by MACCS show no costs associated with accident clean-up, the area immediately around the ship carrying the foreign research reactor spent nuclear fuel (i.e., the dock area) may require some remedial activity).

A sensitivity analysis was performed to address the potential impact of shipboard fires with extremely high temperatures that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum-based fuel or the combustion temperature of the TRIGA fuel. This analysis shows that the maximum consequences of such a fire are a factor of 100 larger than those

Table 4-6 Port Accident Consequences (LCF)

Locations	Accident Severity Category[a] 4	5	6
Elizabeth at the Dock	0.00010	2.8	2.7
Elizabeth in the Channel	0.00016	2.9	2.8
Long Beach at the Dock	0.000093	2.0	2.0
Long Beach in the Channel	0.000035	1.8	1.9
Philadelphia at the Dock	0.000078	1.2	1.2
Philadelphia in the Channel	0.000037	1.2	1.2
Portland at the Dock	0.000034	0.52	0.53
Portland in the Channel	0.000023	0.50	0.51
Norfolk at the Dock	0.000024	0.38	0.37
Norfolk in the Channel	0.000013	0.30	0.30
Charleston at the Dock (Wando Terminal)	0.000011	0.19	0.19
Charleston at the Dock (NWS Charleston)	0.0000068	0.22	0.22
Charleston in the Channel	0.000017	0.19	0.19
Tacoma at the Dock	0.000024	0.75	0.80
Tacoma in the Channel	0.000017	0.63	0.66
Concord NWS at the Dock	0.000019	0.90	0.96
Concord NWS in the Channel	0.000041	1.40	1.50
Jacksonville at the Dock	0.000012	0.31	0.31
Jacksonville in the Channel	0.000011	0.24	0.25
Savannah at the Dock	0.000025	0.23	0.23
Savannah in the Channel	0.0000059	0.18	0.19
Wilmington at the Dock	0.000017	0.22	0.23
Wilmington in the Channel	0.0000042	0.098	0.10
Galveston at the Dock	0.000032	0.64	0.70
Galveston in the Channel	0.000014	0.63	0.69
MOTSU at the Dock	0.0000032	0.099	0.11
MOTSU in the Channel	0.0000042	0.098	0.10

[a] *These accident release categories are the three highest in severity.*

calculated for the base case (Appendix D, Section D.5.4.2.2, Table D-31). An extremely high temperature ship fire is highly unlikely (one in ten billion per shipment) and the risks are comparable to those calculated in the base case. This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 m (1,000 ft). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

Risks

The calculated risk (probability times consequence) to the nearby population on a per-shipment basis assuming one cask per shipment and for the entire basic implementation of Management Alternative 1 is presented in Table 4-7. Each risk value is the sum of the risks from accident severity categories 4, 5, and 6. (A sensitivity study was performed to assess the risks associated with accidents that result in extremely high temperature fires. This sensitivity study was limited to an analysis of the per-shipment risks associated with shipment of spent nuclear fuel through the highest population density port, Elizabeth, NJ. Even though the consequences of this type of an accident are orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case. A more detailed comparison of the base case and sensitivity analyses is presented in Appendix D, Section 5.4.3.2.)

Table 4-7 Port Accident Risks

Port	*Per Shipment of One Cask*		*Total All Shipments*	
	Population Dose (person-rem)	*Population Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Elizabeth via:				
• Two High Population Ports	0.00013	5.6×10^{-8}	0.070	0.000029
• One High and One Intermediate Population Port	0.00011	4.8×10^{-8}	0.060	0.000025
• One High and One Low Population Port	0.00011	4.5×10^{-8}	0.057	0.000024
• Two Intermediate Population Ports	0.000056	2.4×10^{-8}	0.030	0.000013
• One Intermediate and One Low Population Port	0.000051	2.2×10^{-8}	0.027	0.000011
• Two Low Population Ports	0.000046	2.0×10^{-8}	0.024	0.000010
• Direct	0.000042	1.8×10^{-8}	0.022	0.0000094
Long Beach via:				
• Two High Population Ports	0.00011	4.7×10^{-8}	0.058	0.000025
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.038	0.000016
• Two Intermediate Population Ports	0.000050	2.1×10^{-8}	0.026	0.000011
• One Intermediate and One Low Population Port	0.000041	1.8×10^{-8}	0.022	0.0000092
• Two Low Population Ports	0.000032	1.4×10^{-8}	0.017	0.0000072
• Direct	0.000028	1.2×10^{-8}	0.015	0.0000062
Philadelphia via:				
• Two High Population Ports	0.00011	4.5×10^{-8}	0.057	0.000024
• One High and One Intermediate Population Port	0.000088	3.7×10^{-8}	0.047	0.000020
• One High and One Low Population Port	0.000083	3.5×10^{-8}	0.044	0.000019
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.016	0.0000072
• One Intermediate and One Low Population Port	0.000026	1.1×10^{-8}	0.014	0.0000061
• Two Low Population Ports	0.000021	9.3×10^{-9}	0.011	0.0000049
• Direct	0.000017	7.5×10^{-9}	0.0092	0.0000040
Portland via:				
• Two High Population Ports	0.000090	3.8×10^{-8}	0.047	0.000020
• One High and One Intermediate Population Port	0.000059	2.5×10^{-8}	0.031	0.000013
• One High and One Low Population Port	0.000050	2.2×10^{-8}	0.027	0.000011
• Two Intermediate Population Ports	0.000029	1.3×10^{-8}	0.015	0.0000066
• One Intermediate and One Low Population Port	0.000020	9.0×10^{-9}	0.011	0.0000047
• Two Low Population Ports	0.000011	5.1×10^{-9}	0.0059	0.0000026
• Direct	0.0000073	3.2×10^{-9}	0.0039	0.0000017
Norfolk via:				
• Two High Population Ports	0.000095	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000076	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000019	8.3×10^{-9}	0.0098	0.0000044
• One Intermediate and One Low Population Port	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Two Low Population Ports	0.0000088	4.0×10^{-9}	0.0046	0.0000021
• Direct	0.0000048	2.1×10^{-9}	0.0025	0.0000011
Charleston (Wando Terminal) via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.4×10^{-9}	0.0087	0.0000039

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
• One Intermediate and One Low Population Port	0.000012	5.2×10^{-9}	0.0061	0.0000027
• Two Low Population Ports	0.0000066	3.1×10^{-9}	0.0035	0.0000016
• Direct	0.0000027	1.2×10^{-9}	0.0014	6.4×10^{-7}
Charleston (NWS Charleston) via:				
• Two High Population Ports	0.000093	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000017
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000017	7.5×10^{-9}	0.0084	0.0000039
• One Intermediate and One Low Population Port	0.000012	5.3×10^{-9}	0.0058	0.0000028
• Two Low Population Ports	0.0000068	3.2×10^{-9}	0.0032	0.0000016
• Direct	0.0000028	1.3×10^{-9}	0.0011	6.8×10^{-7}
Galveston via:				
• Two High Population Ports	0.000099	4.2×10^{-8}	0.052	0.000022
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000075	3.2×10^{-8}	0.040	0.000017
• Two Intermediate Population Ports	0.000023	1.0×10^{-8}	0.012	0.0000053
• One Intermediate and One Low Population Port	0.000018	8.0×10^{-9}	0.0094	0.0000042
• Two Low Population Ports	0.000013	5.8×10^{-9}	0.0068	0.0000031
• Direct	0.0000090	4.0×10^{-9}	0.0047	0.0000021
Jacksonville via:				
• Two High Population Ports	0.000094	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000075	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000070	2.9×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000018	7.9×10^{-9}	0.0093	0.0000041
• One Intermediate and One Low Population Port	0.000013	5.7×10^{-9}	0.0067	0.0000030
• Two Low Population Ports	0.0000078	3.6×10^{-9}	0.0041	0.0000019
• Direct	0.0000038	1.7×10^{-9}	0.0020	9.0×10^{-7}
Savannah via:				
• Two High Population Ports	0.000093	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000017	7.5×10^{-9}	0.0088	0.0000039
• One Intermediate and One Low Population Port	0.000012	5.3×10^{-9}	0.0062	0.0000028
• Two Low Population Ports	0.0000068	3.2×10^{-9}	0.0036	0.0000017
• Direct	0.0000028	1.3×10^{-9}	0.0015	6.9×10^{-7}
Wilmington via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000073	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000068	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.2×10^{-9}	0.0084	0.0000038
• One Intermediate and One Low Population Port	0.000011	5.0×10^{-9}	0.0058	0.0000026
• Two Low Population Ports	0.0000062	2.9×10^{-9}	0.0032	0.0000015
• Direct	0.0000022	1.0×10^{-9}	0.0012	5.3×10^{-7}
Tacoma via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000062	2.6×10^{-8}	0.033	0.000014
• One High and One Low Population Port	0.000053	2.3×10^{-8}	0.028	0.000012
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.017	0.0000072
• One Intermediate and One Low Population Port	0.000023	1.0×10^{-8}	0.012	0.0000053
• Two Low Population Ports	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Direct	0.0000097	4.3×10^{-9}	0.0051	0.0000023

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
Concord NWS via				
• Two High Population Ports	0.000099	4.2×10^{-8}	0.052	0.000022
• One High and One Intermediate Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• One High and One Low Population Port	0.000060	2.5×10^{-8}	0.032	0.000013
• Two Intermediate Population Ports	0.000038	1.7×10^{-8}	0.020	0.0000087
• One Intermediate and One Low Population Port	0.000029	1.3×10^{-8}	0.016	0.0000067
• Two Low Population Ports	0.000021	9.0×10^{-9}	0.011	0.0000047
• Direct	0.000017	7.1×10^{-9}	0.0088	0.0000038
MOTSU via:				
• Two High Population Ports	0.000091	3.9×10^{-8}	0.048	0.000020
• One High and One Intermediate Population Port	0.000072	3.1×10^{-8}	0.038	0.000016
• One High and One Low Population Port	0.000067	2.8×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000015	6.8×10^{-9}	0.0080	0.0000036
• One Intermediate and One Low Population Port	0.000010	4.6×10^{-9}	0.0054	0.0000024
• Two Low Population Ports	0.0000053	2.5×10^{-9}	0.0028	0.0000013
• Direct	0.0000013	6.2×10^{-10}	0.00069	3.2×10^{-7}

The column on the left indicates the port of entry for the foreign research reactor spent nuclear fuel and the possible combinations of two intermediate ports, as well as no intermediate ports (direct). The second and third columns present two measures of the risk on a per-shipment basis. These risks are based on the conservative assumption that all East and West Coast deliveries would follow the same route for all shipments. The fourth and fifth columns sum the per-shipment risks for all of the shipments, for the entire basic implementation of Management Alternative 1. The two columns under the heading of "Total All Shipments" are the product of the per-shipment risk data for each type of spent nuclear fuel cask with the number of casks of that spent nuclear fuel type. DOE and the Department of State conservatively assumed in these calculations that each port would receive all the casks.

Consider first the per-shipment population exposure risk for a shipment of foreign research reactor spent nuclear fuel to Elizabeth via two high-population density ports. This value, 0.00013 person-rem, is the risk from one cask shipment of the highest nuclear material inventory which would first pass through two high-population density ports, such as Boston and Philadelphia, then would be delivered to Elizabeth. The risk of this cask shipment would be the sum of the risks associated with each of the three ports, because an accident could occur in any of the ports. Since risk is the product of consequences and probability, and probability has no units, the risk would be expressed in the units of the consequences, in this case population exposure (person-rem).

Comparing the risk of sending the foreign research reactor spent nuclear fuel to Elizabeth via two high-population density ports (0.00013 person-rem) to sending the spent nuclear fuel directly to Elizabeth (0.000042 person-rem) shows that the risk would be cut by about two-thirds by eliminating the intermediate ports. This is expected, since the estimated overall risk is the sum of the risk at each of the three ports, and three high-population density ports would have roughly the same risks. Now compare the per-shipment risk of using a low-population density port, say MOTSU, via two low-population density ports. Table 4-7 indicates that this risk would be 0.0000053 person-rem, or about 25 times lower than the highest risk, Elizabeth via two high population ports. All per-shipment risks are conservatively based on the highest nuclear material inventory cask to maximize the potential risk.

The manner of evaluating the per-shipment risk of LCF in Table 4-7 is the same as for the per-shipment population exposure risk. Once again, shipping the foreign research reactor spent nuclear fuel through or into high-population density ports would increase the risk, as would using ships that pass through intermediate ports on their way to the port of entry.

The range of total population risks would be from 0.070 to 0.00069 person-rem for the population dose and from 0.000029 to 3.2×10^{-7} LCF for the risk, comparing shipping to Elizabeth via two high-population density ports and shipping to MOTSU without intermediate ports. The highest estimated population risk due to port accidents that might occur due to the basic implementation of Management Alternative 1 is 0.000029 LCF. This means that there would be less than a one in ten thousand chance of some member of the public incurring an LCF due to the basic implementation of Management Alternative 1 port transits.

The highest estimated MEI accident risk is conservatively determined by multiplying the accident probability by the consequences, in terms of dose to the MEI, of that accident. The MEI in this case is assumed to be an individual at the center of the plume less than 1.6 km (1 mi) from the accident. The highest average MEI doses calculated for the accident severity categories are: 0.11 mrem for category 4, 117 mrem for category 5, and 95 mrem for category 6. See Appendix D, Section D.5.4.2.2 for details. The reason MEI dose for category 6 is relatively lower than that for category 5 is because the larger category 6 associated fire would disperse the radioactive material faster and farther than the category 5 fire. For the 721 shipments in the basic implementation of Management Alternative 1, and using the per port transit accident probabilities in Appendix D, the highest MEI accident risk is estimated to be 0.00042 mrem. This corresponds to about 2×10^{-10} LCF. This means that the chance of the MEI incurring an LCF due to a port accident under the basic implementation of Management Alternative 1 would be less than one in a billion.

Emergency Management and Response

Emergency response capabilities for a foreign research reactor spent nuclear fuel mishap would be available through the U.S. Coast Guard and the local jurisdictions surrounding each candidate port of entry, with specialized support available from DOE. The specialized analysis and identification of potential hazards, use of the robust "Type B" packaging, specific emergency plan and procedure development, training, response rehearsal, and interagency coordination for efficient and effective response would minimize the potential consequences should a foreign research reactor spent nuclear fuel mishap occur. The specific emergency management and response capabilities and responsibilities are described in Chapter 2, Section 2.7.

At military ports, the U.S. Coast Guard routinely provides safety/security screen escorts. The addition of foreign research reactor spent nuclear fuel shipments would have almost no effect on their ongoing operations.

Consequences of Port Accidents

A sensitivity analysis was performed to address the potential impact of extremely high temperature fires, fires that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum based fuel or the combustion temperature of the TRIGA fuel, on the consequences of an accident in port. This analysis, which uses the Port of Elizabeth, NJ as the site of the accident, is presented in Appendix D, Section D.5.4.3.2, and shows that even though the consequences of this type of an accident are two orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case.

This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 meters (1000 feet). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

4.2.2.4 Cumulative Impacts of Port Activities

Port workers are expected to be exposed to other shipments of radioactive materials in addition to those associated with the basic implementation of Management Alternative 1. These shipments include DOE and commercially initiated programs. An assessment has been made of the cumulative impact of the incident-free dose to the maximally exposed worker from all of these activities. The cumulative analysis is based on data collected at several ports for 2.5 years (January 1992 through June 1994). The maximally exposed port worker is estimated to receive less than 10 mrem per year from commercial shipments. Details of this analysis are presented in Appendix D, Section D.4.6. As previously stated, based on cask dose rates equal to the regulatory limit, the maximally exposed port worker could receive an annual dose greater than the NRC and DOE regulatory limit of 100 mrem per year (NRC, 1991). Therefore, DOE would implement mitigation measures.

4.2.2.5 Port Activities Mitigation Measures

As with marine transport, the principal environmental impact that would occur during port activities is radiation dose to workers. No members of the general public would be close enough to the transportation cask to receive any radiation dose. The workers would receive this dose during safety inspections and handling activities which cannot be curtailed.

Two conservative assumptions in this analysis drive the maximally exposed worker dose higher than would actually be expected. The radiation dose rate near every foreign research reactor spent nuclear fuel shipping container is assumed to be equal to the regulatory limit and the same individual is assumed to conduct all the inspections. Neither of these is actually likely to occur.

Nevertheless, DOE and the Department of State would require, through a clause in the shipping contracts, some administrative controls on the port workers to mitigate the radiation doses to the workers during inspection and handling activities. DOE and the Department of State would implement a system to track the inspectors and other port workers actually involved in the shipment of foreign research reactor spent nuclear fuel. If any inspector's or worker's dose approaches 100 mrem in any year, then DOE and the Department of State would require other inspectors or workers to be used. In this way, the maximally exposed worker dose would be constrained to the regulatory limit.

If a cask or casks were sunk in coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2.6 Environmental Justice at the Port(s)

Executive Order 12898 deals with the issue of environmental justice and directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations.

The concept of environmental justice is discussed in more detail in Appendix A. During normal port activities associated with receipt of the foreign research reactor spent nuclear fuel shipments—including harbor activities, unloading the ship, transfer of the spent nuclear fuel containers to truck or train, and movement out of the port city—the dominant radiological impacts have been shown to be the exposures received by the workers in the immediate vicinity of the shipping container. These individuals include the inspectors, shipping container handlers, truck drivers, etc. Since the intensity of the gamma radiation falls off rapidly with distance, the doses that might be received by other workers and members of the general population can in theory be calculated, but would not generally be measurable or distinguishable from natural background radiation.

Potential radiological impacts to people residing near the port are associated with low probability (less than one in a million) accidents that are so severe that the spent nuclear fuel casks would be ruptured and a fire would burn long enough around the cask that some of the radioactive material would be released. In this case, some of the radioactive spent nuclear fuel might be vaporized and lifted by the heat of the fire and carried downwind of the accident location. Where and how far this radioactive material would go before being deposited on the ground would depend on how high the heat from the fire lofts it and the particular weather conditions at the time. Most of this vaporized spent nuclear fuel would be expected to be deposited in the first few kilometers downwind of the fire but small amounts could be carried out for several tens of kilometers.

Because the particular details of both the accident conditions (such as the severity of a fire) and the weather conditions at the time of an accident could vary so much, a range of accident conditions and wind directions, wind speeds, and other weather conditions were examined during the evaluation of accidents (see Section 4.2.2.3). Population impact evaluations were performed for distances out to 80 km (50 mi). The risk of LCF was found to be so small that zero LCF would be expected due to accidents at ports.

Appendix A describes minority populations and low-income households residing near the ports. Calculations for incident-free and accident conditions clearly demonstrate that for the general population, including minority and low-income groups, the radiological impacts would be very low. Minority or low-income populations living near the potential ports of entry would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to the same very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at candidate ports, including the social and economic status of the general population, minority populations, and the low-income population surrounding candidate ports. Economic benefits that would result from increased cargo handling and transportation in the port area would be extremely small for the general population or any particular segment of the population residing near candidate ports.

4.2.3 Ground Transport Impacts

Foreign research reactor spent nuclear fuel is transported in large, heavy containers called transportation casks. Transportation casks are designed and constructed to contain the radioactivity in spent nuclear fuel during severe transportation accidents. NRC has estimated that transportation casks will withstand 99.4 percent of truck and rail accidents without sustaining damage sufficient to breach the transportation cask (NRC, 1987). Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport spent nuclear fuel be so damaged that there is a reasonable possibility of release of radioactivity to the environment.

Spent nuclear fuel has been transported along highways, railways, and waterways since 1949. Federal standards describe the routing requirements for spent nuclear fuel shipments. Spent nuclear fuel transported includes foreign research reactor, commercial, naval, and DOE spent fuel. Since 1949, there have been 21 incidents involving vehicles carrying irradiated fuel elements. None of these incidents resulted in damage to the structural integrity of a cask or the release of the cask's contents.

4.2.3.1 Conservative Assumptions and Analytic Approach

Transportation impacts may be divided into two parts: the impacts due to incident-free transportation and the impacts due to transportation accidents. For incident-free transportation and transportation accidents, impacts may be further divided into two parts: nonradiological impacts and radiological impacts. The nonradiological impacts consist of the vehicular impacts of transportation, such as vehicular emissions and traffic accidents.

For incident-free transportation, the radiological impacts would result from the radiation field that surrounds the cask. For transportation accidents, the radiological impacts would be based on the radioactivity released from the spent nuclear fuel transportation cask during the accident. Impacts are estimated for workers and the population along the transportation route.

For both incident-free transportation and transportation accidents, methodology developed by NRC and used by DOE in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (SNF&INEL Final EIS) (DOE, 1995c) was used to estimate the impacts for foreign research reactor spent nuclear fuel in this EIS. These impacts were quantified as the estimated number of radiation-related cancer fatalities and the estimated number of nonradiological fatalities from vehicular emissions and traffic accidents. Appendices B, C, D, E, and F of this EIS contain more details on the equipment, regulations, and experience associated with spent nuclear fuel transportation, and the methodology, data, and conservative assumptions used to develop these estimates.

Under the basic implementation of Management Alternative 1, acceptance of the foreign research reactor spent nuclear fuel would require the transport of approximately 837 casks from seaports and Canadian border crossings to DOE facilities. The number of casks was determined by assigning the spent nuclear fuel from each foreign research reactor to a reasonably available and capable cask. Conservative assumptions were used to estimate cask capacity, which is based on physical, thermal, and radiological characteristics of the spent nuclear fuel. Appendix B contains more details on the foreign research reactor spent nuclear fuel transportation casks.

For the purposes of analysis in this EIS, the initial ground transportation activities to the Savannah River Site and/or the Idaho National Engineering Laboratory is called Phase 1, and the possible subsequent intersite ground transport and continued management is called Phase 2. The impact assessment includes analysis of between 13 and 161 intersite shipments, depending upon the mode of transportation (truck or rail) and the potential foreign research reactor spent nuclear fuel management sites that might be selected. Intersite shipments would be fewer than foreign shipments because the spent nuclear fuel would be cooler. Larger casks would likely be used, and more foreign research reactor spent nuclear fuel would be consolidated.

The first step in the ground transportation analysis was to determine the incident-free and accident risk factors, on a per-shipment basis assuming one cask per shipment, for transportation of the various spent nuclear fuel casks. Risk factors, as any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities and the magnitudes of exposure are discussed in Appendix E. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the cask, and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure and the magnitudes of exposure are discussed in Appendix E.

Calculation of risk factors was accomplished by first using the HIGHWAY (Johnson, et al., 1993a) and INTERLINE (Johnson, et al., 1993b) computer codes to choose representative routes in accordance with the U.S. Department of Transportation regulations. These codes provide population estimates along the routes so that the RADTRAN (Neuhauser, 1993) and RISKIND (Yuan, et al., 1993) codes could be used to determine the risk factors associated with ground transportation activities. These computer codes are described in more detail in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and Appendix E of this EIS.

The single largest contributor to the ground transport population doses (about 80 percent) calculated with RADTRAN was found to be the dose to members of the public at truck stops. The parameters used to calculate doses during truck stops are quite conservative. The parameters are based on the assumption that stops occur as a function of distance, with a truck stop rate of 0.011 hr per km (0.018 hr per mi). This stop rate results in over an hour of stop time per 100 km (62 mi) of travel. It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m (66 ft). These parameters were used because they are the default parameters in the RADTRAN code and they were used in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These conservative assumptions that are built in the code are highly unlikely to occur.

The next step is to use the risk factors and the number of shipments to estimate the risk of every possible way the foreign research reactor spent nuclear fuel program could be implemented. Because of the large number of ports, cask types, spent nuclear fuel types, and implementation options, simplifying assumptions are needed to control the amount of repetitive analysis:

- A review of the accident risk factors for the various types of spent nuclear fuel (see Appendix E) indicates that there is relatively little variation between the different types of foreign research reactor spent nuclear fuel, thus, it is not overly conservative to use the highest risk factors for all shipments.

- Spent nuclear fuel from countries bordering the Atlantic Ocean and Mediterranean Sea was assumed to arrive on the East Coast of the United States. Spent nuclear fuel from countries bordering the Indian and Pacific Oceans was assumed to arrive on the West Coast. This is conservative from an overland transportation standpoint, because, as shown in Appendix E, marine shipment to the coast nearest the management site would reduce the risk factors for the overland shipment.

- To account for the return transport of empty casks, the impacts due to vehicle emissions and traffic accidents were multiplied by two.

The foreign research reactor spent nuclear fuel could arrive at any of the ports of entry selected by DOE and the Department of State using criteria that are detailed in Appendix D, and would be likely to arrive at a variety of these ports. Therefore, the proposed impacts were completely analyzed three times, consisting of an upper bounding case, a lower bounding case, and an average case, for both truck and rail shipments. The upper bound case conservatively assumes the port(s) with the highest risk factors was chosen for each transportation activity. The risk factors are generally a function of distance and total population along the port to management site route, so the port chosen often shifted between Phase 1 and Phase 2. Conversely, the lower-bound case assumes ports with the lowest risk factors.

The average case is designed to provide a realistic estimate of the ground transport risk of transporting the foreign research reactor spent nuclear fuel. The risk factors are an arithmetic average of the risk factors for all acceptable ports. This represents the risk associated with the basic implementation of Management Alternative 1 and receiving foreign research reactor spent nuclear fuel at a variety of commercial ports.

Since each potential port of entry and each management site is capable of receiving spent nuclear fuel via rail or highway, the program was analyzed using each mode of transportation. The exception to this is the Nevada Test Site which has no existing rail capability, so that link was approximated by a hypothetical rail line to the Yucca Mountain Site. Additionally, the potential to use trucks to carry the relatively small casks from ports to potential foreign research reactor spent nuclear fuel management sites and rail to carry larger casks between potential foreign research reactor spent nuclear fuel management sites was analyzed. Site to site shipment would not occur until approximately 2006, so it is difficult to precisely predict which cask would be used. The analysis is based on a truck cask that carries 4 times as much spent nuclear fuel as a foreign cask, and a rail cask that carries 10 times as much spent nuclear fuel.

4.2.3.2 Impacts of Incident-Free Ground Transport

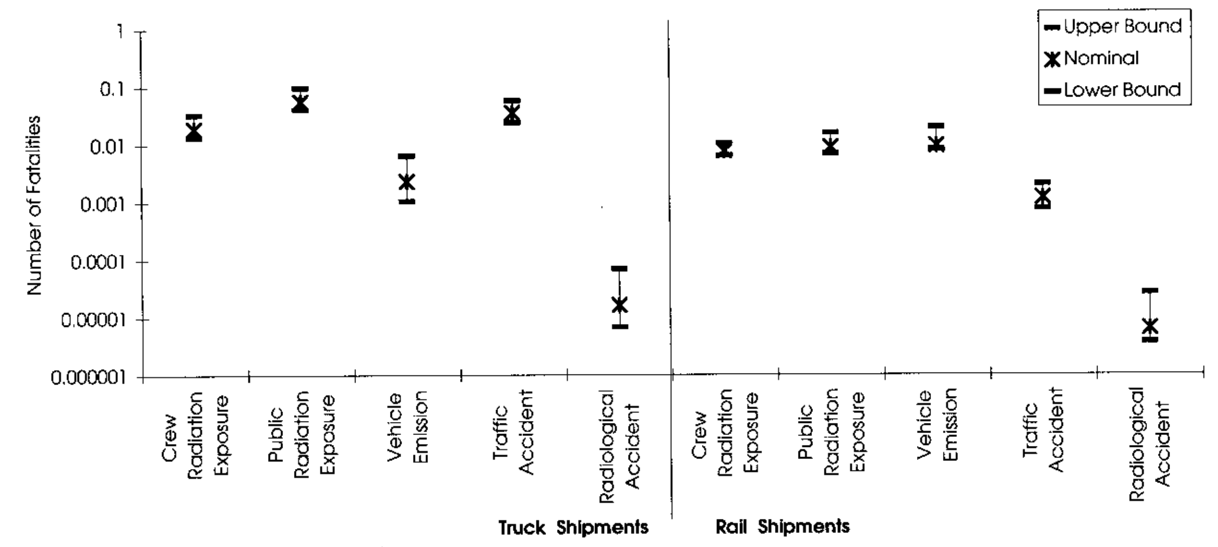
The incident-free transportation of spent nuclear fuel was estimated to result in total population risk that ranged from 0.013 to 0.30 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the transportation workers. Thus, the calculated maximum risk value for overland transportation is less than one fatality from cancer due to the basic implementation of Management Alternative 1. The range of fatality estimates is caused by two factors: (1) the option of using truck or rail to transport spent nuclear fuel; and (2) combinations of Phase 1 and Phase 2 sites that create varying cask shipment numbers and distances.

The estimated number of LCF due to radiation exposure for transportation workers ranged from 0.006 to 0.071. The estimated number of radiation-related LCF for the general population ranged from 0.007 to 0.22, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.052. These incident-free results apply to the workers and the public because both would be close enough to the cask to receive some radiation dose.

The impacts of transportation which are based on four Programmatic SNF&INEL Final EIS (DOE, 1995c) programmatic alternatives are summarized in Figures 4-2 through 4-5. The impacts of these additional programmatic alternatives are described in more detail in Appendix E.

The highest estimated ground transport maximally exposed worker risk is 0.00052 LCF, just like the marine transport and port worker risks. This estimate is based on the conservative assumption that one truck driver makes enough trips to reach the regulatory limit of 100 mrem per year every year for 13 years. This means that under the assumptions described above, the chance of this individual incurring an LCF due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

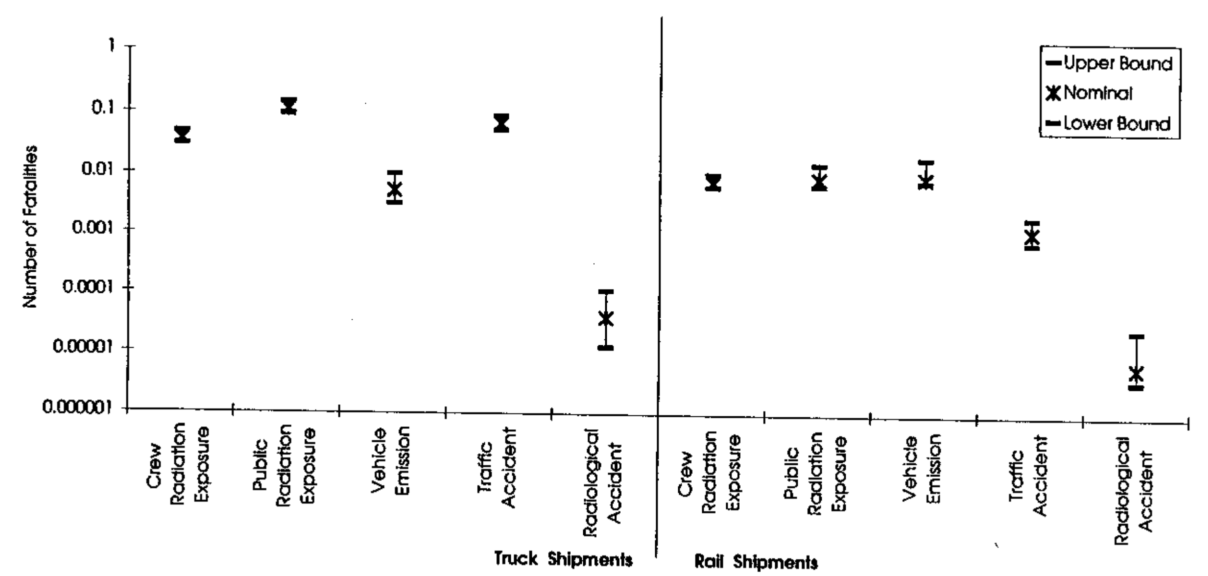
The highest estimated incident-free population risk is 0.30 LCF, which means that there would be a 30 percent chance of one additional cancer fatality among the public and the ground transport workers due to the basic implementation of Management Alternative 1.



- Analysis of Decentralization applies to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives

- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-2 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-3 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

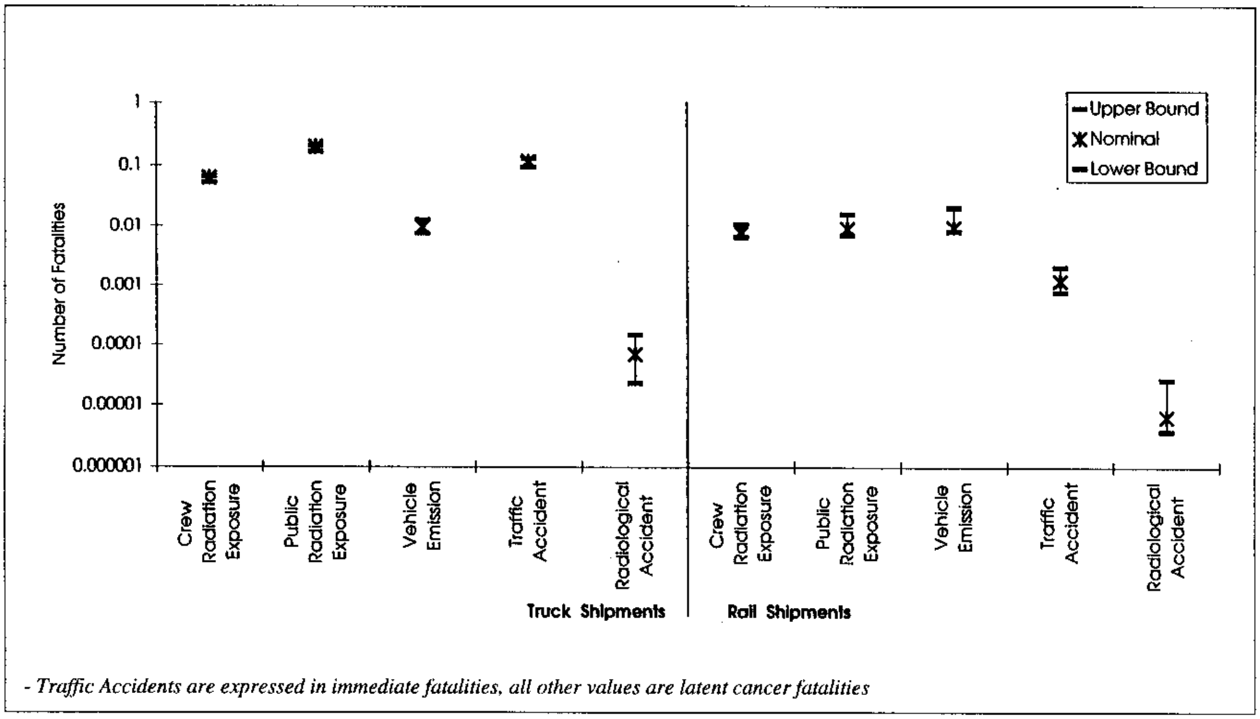


Figure 4-4 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

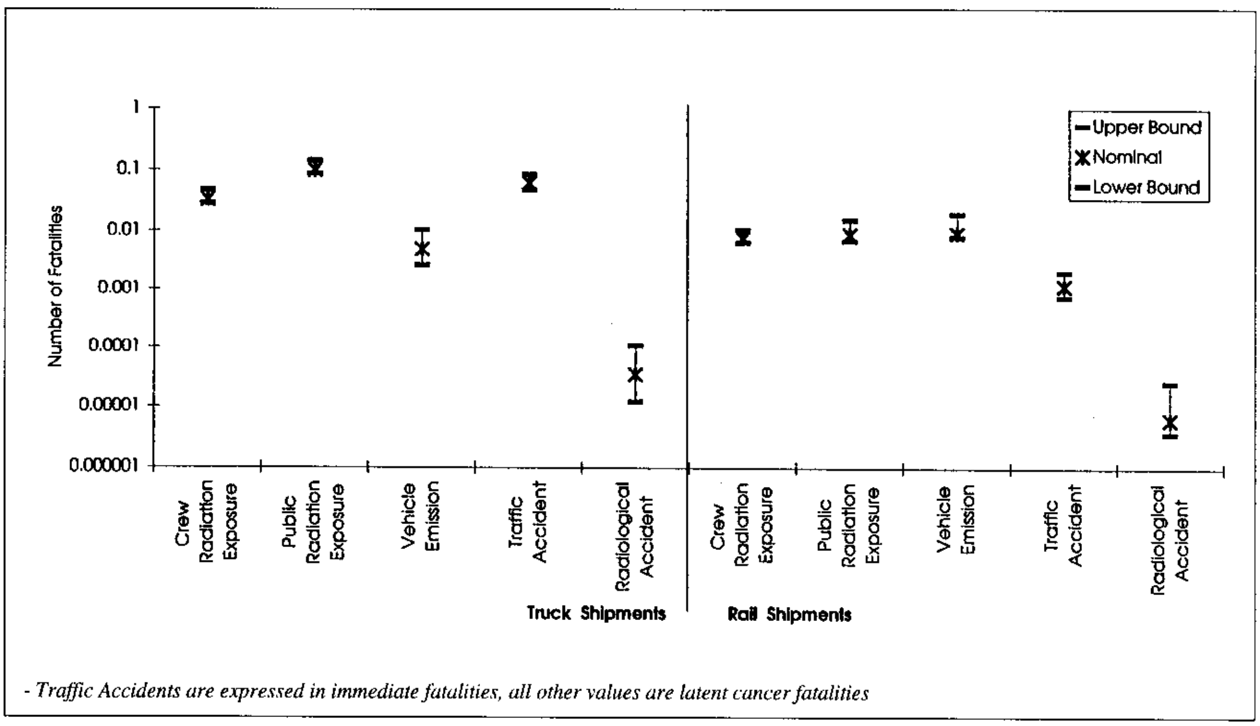


Figure 4-5 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

4.2.3.3 Impacts of Accidents During Ground Transport

The most severe accidents that might reasonably occur on this leg of the journey are truck or train crashes, followed by a large fire. If an accident occurred on a causeway at or near a port that caused a cask to fall into seawater, the consequences would be the same as if a cask fell off a ship into seawater. These consequences are presented in Section 4.2.1.3 under the subheading "Sunken Cask." Each State, and most local jurisdictions, maintain a hazardous materials response capability and a radiological protection program. These capabilities, along with the DOE radiological response assets that would be on-call for immediate technical assistance and response, would provide a high-level of expertise and would reduce the potential impacts of a foreign research reactor spent nuclear fuel accident.

Since hazardous materials team training is required to include radiological materials response, each team possesses a basic level of understanding and capability for a foreign research reactor spent nuclear fuel incident response. An incremental enhancement for spent nuclear fuel-specific response characteristics and planning may be required, especially for those jurisdictions along selected routes whose emergency responders are primarily volunteer organizations.

The development of a transportation plan specifically for the shipping campaign that would incorporate and integrate State and local emergency response plans, would increase emergency responder effectiveness and reduce the potential consequences of a foreign research reactor spent nuclear fuel accident.

Each State's emergency planning infrastructure, using the Local Emergency Planning Committees to the State Emergency Response Commission, enables these jurisdictions to identify and resolve potential emergency management and response issues and communicate issues that would require DOE and Department of State attention. This, along with DOE's Transportation External Coordination/Working Group, would ensure that all concerned agencies would be involved in the planning process to address potential problems before they become major hazards.

Risks

The total ground transportation accident risks for the basic implementation of Management Alternative 1 are estimated to range from 0.000004 to 0.00028 LCF from radiation and from 0.001 to 0.14 for traffic fatality, depending on the transportation mode and potential foreign research reactor spent nuclear fuel management sites that might be selected. Section 4.10 compares these risks to those of common activities. The reason for the range of fatality estimates is the same as those described for incident-free transportation. The risk of 0.14 for a traffic fatality means that under these conservative assumptions there would be a 14 percent chance of a traffic fatality related to the basic implementation of Management Alternative 1.

The maximum foreseeable offsite transportation accident would involve a shipment of foreign research reactor spent nuclear fuel in a suburban population zone under neutral (average) weather conditions. The accident has a probability of occurrence of about 0.0000001 per year (one chance in ten million), and could result in 14 person-rem and no fatalities. The probability of an accident occurring is at least an order of magnitude smaller in either an urban area or under stable atmospheric conditions. The consequences are less than an order of magnitude larger.

The impacts of transportation accidents are summarized in Figures 4-2 through 4-5, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risk under each representative Programmatic SNF&INEL Final EIS alternative.

The highest estimated MEI radiological risk to members of the public due to accidents during ground transport is 1.4×10^{-11} LCF. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in ten billion.

The highest estimated population radiological risk due to accidents is 0.00028 LCF, which is much less than one LCF.

4.2.3.4 Ground Transport Cumulative Impacts

The Programmatic SNF&INEL Final EIS (DOE, 1995c) analyzed the cumulative impacts of ground transportation, taking into account impacts from: (1) historical shipments of spent nuclear fuel to the five proposed foreign research reactor spent nuclear fuel management sites; (2) the programmatic alternatives; (3) other reasonably foreseeable actions that include transportation of radioactive material; and (4) general radioactive materials transportation that is not related to a particular action. The transportation of foreign research reactor spent nuclear fuel is included in the calculated totals under the spent nuclear fuel shipments for the Programmatic SNF&INEL Final EIS Alternatives 1 through 5. Proposed transportation of all spent nuclear fuel (of which the foreign research reactor fuel is a small component) accounts for less than one percent of the total LCF attributable to the transportation of radioactive material, and foreign research reactor spent nuclear fuel accounts for less than one quarter of that one percent. The total number of LCF over the time period 1943 through 2035 was estimated to be 290.

4.2.3.5 Ground Transport Mitigation Measures

The principal environmental impacts that would occur during ground transport are: (1) LCF due to radiation exposure, (2) LCF due to vehicular emissions, and (3) immediate fatalities due to traffic accidents. All three of these would be reduced by choosing port(s) of entry close to the management site(s). This would minimize the distance that must be covered by the vehicle(s).

Furthermore, in the case of truck transport, the truck driver(s) would be monitored for radiation dose. The annual maximally exposed worker limit of 100 mrem would never be approached during any single shipment, but the same driver could be used for multiple shipments throughout a year. DOE would implement mitigation measures through the foreign research reactor spent nuclear fuel acceptance contracts to ensure that each individual driver's dose remains below the regulatory limit. If any individual truck driver accumulates a dose approaching this limit in a year, DOE would require that new driver(s) be used to keep each individual driver's dose below the regulatory limit.

Since the casks would produce a radiation field of less than 10 mrem/hr at 2 m (6.6 ft) from the vehicle, an individual member of the general public would have to be within 2 m (6.6 ft) of the vehicle for at least ten hours in a year to receive a dose equal to the regulatory limit of 100 mrem/yr. A truck is not likely to sit in a traffic jam right beside another vehicle for as long as ten hours and an individual gas station attendant is not likely to spend ten hours refueling the trucks carrying foreign research reactor spent nuclear fuel. Therefore, DOE does not plan to implement ground transport mitigation measures for members of the general public.

4.2.3.6 Barge Transport

DOE and the Department of State have examined the possibility of using barges for the transport of foreign research reactor spent nuclear fuel as a substitute for truck or rail transport. The only two locations where barge transport is feasible are from the Port of Portland, OR up the Columbia River to the Hanford

Site and from the Port of Savannah, GA up the Savannah River to the Savannah River Site. Barge transport could only be implemented if one or both of these port/site combinations is selected in the Record of Decision.

For barge transport up the Columbia River, the incident-free radiological risk to the public would be approximately 0.0000043 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000029 and 0.0000058 LCF per shipment, respectively. For barge transport up the Savannah River, the incident-free radiological risk to the public would be approximately 0.0000019 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000028 and 0.0000026 LCF per shipment, respectively.

For barge transport up the Columbia River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 3.5×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 1.5×10^{-8} and 3.8×10^{-9} LCF per shipment, respectively. For barge transport up the Savannah River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 2.9×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 9.4×10^{-9} and 1.1×10^{-9} LCF per shipment, respectively.

The barge transport analysis is presented in more detail in Appendix E, Section E.8.15. The net result is that the foreign research reactor spent nuclear fuel could be transported by barge with approximately the same level of risk to workers and the public as if it was transported by truck or rail.

4.2.3.7 Environmental Justice Along Ground Transport Routes

The dominant radiological risks and impacts associated with incident-free transportation activities are the exposures received by the workers in the immediate vicinity of the casks and people who might be near the casks at truck stops. These individuals would be the only people receiving a measurable exposure during a spent nuclear fuel shipment. As discussed in Section 4.2.3.2, the number of radiation-related latent cancer deaths among transportation workers and the general public combined was calculated to be less than one. The same is true for cancer due to vehicle emissions. Ground transportation accidents would be expected to result in no additional radiological impacts to the population in the vicinity of the accident. Potential impacts from low probability accidents vary considerably and are dependent on the accident conditions (such as the size of the resulting fire, if any) and the weather conditions at the time of an accident. Transportation accidents were estimated to result in no LCF due to radiation and less than 0.2 immediate deaths due to traffic fatalities (see Section 4.2.3.3).

As described in Appendix A, the percentage of the total population comprised of minorities or low-income households varies among routes. Calculations for incident-free and accident conditions demonstrate that for the general population the radiological impacts would be very low. Minority or low-income populations living near these routes would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment along transportation routes, including the social and economic status of the general population, minority populations, and the low-income population residing along the transportation routes. Economic benefits that would result from increased transportation of cargo along transportation routes would be extremely small for the general population or any particular segment of the population residing along the transportation routes.

4.2.4 Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section presents the potential environmental impacts from the basic implementation of Management Alternative 1 at the potential foreign research reactor spent nuclear fuel management sites, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. It summarizes the detailed site analysis presented in Appendix F, Sections F.4, F.5, and F.6. The analysis examined environmental topics such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational and public health and safety, noise, traffic and transportation, utilities and energy, and waste management. The analysis showed that the basic implementation of Management Alternative 1 would not have a major effect on any of the environmental topics. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

Because of the public interest in radiation exposure to workers and the public, Section 4.2.4.1 discusses in detail the impacts on occupational and public health and safety from the basic implementation of Management Alternative 1, even though the analysis concludes that such impacts are very low. Section 4.2.4.2 summarizes the impacts on the other environmental topics. Section 4.2.4.3 discusses the cumulative impacts of the basic implementation of Management Alternative 1 at each candidate management site, and Section 4.2.4.4 addresses the waste management and mitigation measures available under the basic implementation of Management Alternative 1. Later in this chapter, Section 4.10 compares the risks of the basic implementation of Management Alternative 1 to risks of common activities.

4.2.4.1 Occupational and Public Health and Safety

Possible sources of occupational and public radiological exposure from foreign research reactor spent nuclear fuel include: (1) emissions of radioactive material from incident-free operations, (2) incident-free handling activities, and (3) emissions from accident conditions. Foreign research reactor spent nuclear fuel management is not expected to impact occupational and public health and safety. Nonradiological exposures are not likely to occur during construction or operation of foreign research reactor spent nuclear fuel storage facilities. Radiological exposures are presented in individual subsections for emissions-related impacts, handling-related impacts, and accident-related impacts.

Conservative Assumptions and Impacts to the Public of Incident-Free Site Activities

Doses that could be received by the public during incident-free operation of foreign research reactor spent nuclear fuel storage facilities could only be due to emissions of radioactive material that becomes airborne. The public would be too far from the storage facilities to receive any direct exposure. In summary:

- Doses were calculated for the MEI, defined as an individual living at the management site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions released during foreign research reactor spent nuclear fuel transfer from the transportation cask to the storage facility and from foreign research reactor spent nuclear fuel storage.

- Radiological airborne emissions consist of two parts: (1) emissions from gaseous releases during receipt and unloading of the transportation casks; and (2) emissions during the management period. The emissions during receipt and unloading were calculated conservatively assuming one percent of the foreign research reactor spent nuclear fuel would fail during transport and the associated gaseous fission products would be released during the transfer at the management site. DOE and the Department of State also conservatively assumed that unloading the spent nuclear fuel cask in a dry cell would allow all free gaseous fission products to be released to the environment, while unloading in a wet pool would allow 90 percent of the halogens to be retained in the water. Radiological emissions during wet storage were based on historical data at the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Site. The emissions during incident-free dry storage would be zero because the spent nuclear fuel would be stored in sealed containers. The methodology and conservative assumptions used for the calculation of radiological emissions under the basic implementation of Management Alternative 1 are discussed in detail in Appendix F, Section F.6.

- Doses were calculated separately for each phase of the program at each candidate management site to accommodate the two-phased implementation of the basic implementation of Management Alternative 1. For example, in the case where the Nevada Test Site, the Hanford Site, or the Oak Ridge Reservation is selected as a Phase 2 site, with the Savannah River Site or the Idaho National Engineering Laboratory as a Phase 1 site, doses were calculated at the Savannah River Site or the Idaho National Engineering Laboratory for Phase 1, and at the Hanford Site, Oak Ridge Reservation, or the Nevada Test Site for Phase 2.

- Doses from an operation which combines an existing wet or dry storage facility for spent nuclear fuel receiving and characterization and dry storage casks to enhance storage capacity are bounded by the doses calculated for the existing facility.

- Doses were conservatively calculated for the maximum quantity of foreign research reactor spent nuclear fuel that could be received at each storage site as discussed in Appendix F, Section F.4.

Tables 4-8 through 4-12 summarize the annual emission-related doses to the public and the associated risks for the MEI and population at each site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period. In general, receipt and unloading at wet storage facilities produces lower public risk than at dry storage facilities.

Table 4-8 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Savannah River Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet storage)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet storage)	0.000073	3.7×10^{-11}	0.0046	0.0000023
• New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at:				
• RBOF (wet storage)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet storage)[a]	0.00036	1.8×10^{-10}	0.022	0.000011
• New Dry Storage Facility	0	0	0	0

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Table 4-9 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF[a]/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• Fluorinel Dissolution and Fuel Storage (FAST) (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
• New Dry Storage Facility[b]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at:				
• IFSF[a]/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
• New Dry Storage Facility[b]	0	0	0	0

[a] *Irradiated Fuel Storage Facility*

[b] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Table 4-10 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Hanford Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Fuel Material Examination Facility (FMEF) (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility[a]	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are different from those for FMEF due to the different release height and location.*

Table 4-11 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Table 4-12 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Engine Maintenance and Disassembly (E-MAD) (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
• New Dry Storage Facility[a]	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for E-MAD.*

Among all the potential foreign research reactor spent nuclear fuel management sites, the maximum estimated annual incident-free public MEI radiological exposure from emissions is 0.09 mrem per year. This exposure would occur at the Oak Ridge Reservation (Table 4-11) during receipt and handling. It is much higher than all other corresponding dose rates in Tables 4-8 through 4-12. The receipt period would be about 3 years, so the total MEI dose would be 0.27 mrem. The associated probability for incurring one LCF would be 1.4×10^{-7} for the MEI, which represents less than two chances in ten million of developing a fatal cancer from radiological exposure.

The highest annual incident-free population risk among the Savannah River Site and the Idaho National Engineering Laboratory (Phase 1 sites) is 0.000011 LCF per year (Tables 4-8 and 4-9), which would be due to emissions from L-Reactor Basin at the Savannah River Site. Assuming some foreign research reactor spent nuclear fuel is stored in this basin for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be as high as 0.00014 LCF. The highest annual incident-free population risk from a new dry storage facility at a potential Phase 2 site (Tables 4-8 through 4-12), is 0.000043 LCF per year, which would be due to receipt/unloading at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.00013 LCF. This is higher than any other combination of Phase 2 dry storage annual risks and durations. Adding the Phase 1 and Phase 2 population risks yields 0.00027 LCF for the total population risk to the public living near the sites due to incident-free conditions.

Conservative Assumptions and Impacts to Workers of Incident-Free Site Activities

Workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the management site, transferring foreign research reactor spent nuclear fuel from one facility to another within the management site, or packaging the foreign research reactor spent nuclear fuel for shipment to another management site. Detailed analysis of the potential impacts is given in Appendix F of this EIS. In summary:

- The maximally exposed worker dose estimate is based on the regulatory limit of 5,000 mrem per year for radiation workers at all DOE management sites. DOE and the Department of State conservatively assumed that an individual worker received this dose every year for all 13 years that the handling operations would be in progress. Although this assumption is highly unlikely, the calculated total maximally exposed worker dose is 65,000 mrem and the associated risk is 0.026 LCF. This means that this individual would have a nearly three percent higher chance of incurring an LCF.

- Worker population doses were estimated by considering the type and duration of all operations performed by the workers during the handling of each transportation cask and storage cask as appropriate, including: (1) the number of workers needed, (2) the duration of a specific operation, and (3) the distance between the transportation cask and the operation being performed. Only the workers actually performing the operations receive radiation doses, and thus would have an increased risk of incurring an LCF. If the total radiation dose is received by a small number of workers, each worker would have a higher risk of cancer than if the total dose is received by a large number of workers. The dose rate in the vicinity of the transportation and storage casks assumed for the estimates was based on the conservative methodology presented in Appendix F, Section F.5. As noted in Section F.5, worker population doses associated with dry storage cask design may be higher than those associated with the vault design because of the additional worker activities associated with the handling of the cask that transfers the canistered spent fuel to the concrete structure. The worker population doses reported below for new dry storage conservatively reflect the cask design.

- The number of casks handled at each potential foreign research reactor spent nuclear fuel management site would depend on the number of cask shipments considered under the ground transportation options discussed in Section 2.6.4.1, and the amount of foreign research reactor spent nuclear fuel expected to be transferred between facilities during Phase 2.

Table 4-13 provides a summary of the number of casks that would be handled at each potential foreign research reactor spent nuclear fuel management site under the Centralization Alternative in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and in the current EIS.

Table 4-13 Estimated Number of Shipments to and from Each Potential Foreign Research Reactor Spent Nuclear Fuel Management Site

Candidate Storage Site	*Incoming Shipments*	*Intersite Shipments*	*Outgoing Shipments*	*Total Shipments*
Savannah River Site or Idaho National Engineering Laboratory Phase 1	644[a]	0	161	805
Savannah River Site or Idaho National Engineering Laboratory Phases 1 and 2	837[b]	209	0	1,046
Hanford Site or Oak Ridge Reservation or Nevada Test Site Phase 2	354[c]	0	0	354

[a] *10-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[b] *13-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[c] *161 from near term site using large truck casks and 193 from ports using foreign research reactor spent nuclear fuel certified casks.*

Tables 4-14 through 4-18 present the population doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at each management site. The results do not include shipments in large rail casks.

Table 4-14 Handling-Related Impacts to Workers at the Savannah River Site

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	RBOF/L-Reactor	*New Dry Storage*	*RBOF/L-Reactor*	*New Dry Storage*
Phase 1	250	NA	0.10	NA
Phases 1 and 2	NA	416[a]	NA	0.17[a]

[a] *Cask design*

Table 4-15 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory

	Worker Population Dose (person-rem)			*Worker Population Risk (LCF)*		
	IFSF[a]/CPP-749	*FAST*	*New Dry Storage*	*IFSF[a]/CPP-749*	*FAST*	*New Dry Storage*
Phase 1	257	250	NA	0.10	0.10	NA
Phases 1 and 2[b]	NA	NA	424[c]	NA	NA	0.17[c]
Phases 1 and 2[d]	NA	NA	416[c]	NA	NA	0.17[c]

[a] *Irradiated Fuel Storage Facility*

[b] *Phase 1 at IFSF/CPP-749*

[c] *Cask design*

[d] *Phase 1 at FAST*

Table 4-16 Handling-Related Impacts to Workers at the Hanford Site

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	FMEF/New Dry Storage	*FMEF/New Dry Storage*
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-17 Handling-Related Impacts to Workers at the Oak Ridge Reservation

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Dry Storage	*New Dry Storage*
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-18 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	E-MAD	*New Dry Storage*	*E-MAD*	*New Dry Storage*
Phase 2	113	266[a]	0.05	0.11[a]

[a] *Cask design*

According to the above tables, the highest dose to a working crew at a single site would be 424 person-rem at the Idaho National Engineering Laboratory in the analyzed case which assumes that all foreign research reactor spent nuclear fuel is received in the Irradiated Fuel Storage Facility and/or the CPP-749 facility (dry storage) during Phase 1 and is transferred to a new dry storage facility at the Idaho National Engineering Laboratory in Phase 2. The associated number of additional LCF is 0.17. The highest dose to working crews for both phases in more than one site is 523 person-rem: 266 person-rem at one of the 3 Phase 2 sites, plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.21.

Conservative Assumptions and Accident-Related Impacts

An evaluation of hypothetical accidental radioactive material releases at the potential foreign research reactor spent nuclear fuel management sites was performed to assess the impact of possible radiation exposure to individuals and the general population (see also Appendix F, Section F.6). All inputs are site-specific except for the radioactivity release. Site-specific information includes meteorological conditions, population distribution, and food production and consumption rates within 80 km (50 mi) of the management location.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel management facility:

- Worker: An individual located 100 m (330 ft) from the radioactive material release point. (The impact of accidents on close-in workers is not calculated numerically but is discussed qualitatively for each accident at the end of this section.) For elevated release (from a tall stack), the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release.

The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- Maximally Exposed Individual (MEI): A theoretical member of the general public living at the management site boundary receiving the maximum exposure. This individual is conservatively assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the conservative 95th-percentile meteorological condition.

- Nearest Public Access Individual (NPAI): An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the conservative 95th-percentile meteorological condition.

- General population within an 80-km (50-mi) radius of the facility: The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the conservative 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume the contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeded the protective action guidelines developed by the U.S. Environmental Protection Agency (EPA, 1991a). To ensure a consistent and conservative analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those that were analyzed in the Programmatic SNF&INEL Final EIS. The selection of the accidents was based on the following considerations:

(1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent fuel element); and (3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each management location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a spent nuclear fuel cask, and an aircraft crash with an ensuing fire.

Tables 4-19 through 4-23 present the frequencies and the consequences of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the conservative assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE and the Department of State did not estimate the worker population dose due to accidents.

Table 4-19 Frequency and Consequences of Accidents at the Savannah River Site

		Consequences							
		MEI		*NPAI*		*Population*		*Worker*	
	Frequency (per yr)	*(mrem)*	*(LCF)*	*(mrem)*	*(LCF)*	*(person-rem)*	*(LCF)*	*(mrem)*	*(LCF)*
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	0.24	$1.2x10^{-7}$	0.068	$3.4x10^{-8}$	9.2	0.0046	28	0.000011
Dropped Spent Nuclear Fuel Cask	0.0001	0.018	$9.0x10^{-9}$	0.00034	$1.7x10^{-10}$	0.55	0.00028	0.28	$1.1x10^{-7}$
Aircraft Crash w/Fire	$1x10^{-6}$	40	0.00002	0.29	$1.5x10^{-7}$	1300	0.65	120	0.000048
Wet Storage Accidents - RBOF									
Spent Nuclear Fuel Assembly Breach	0.16	0.0070	$3.5x10^{-9}$	0.00039	$2.0x10^{-10}$	0.23	0.00012	0.14	$5.6x10^{-8}$
Accidental Criticality	0.0031	130	0.000065	44	0.000022	4,800	2.4	16,000	0.0064
Aircraft Crash	$1x10^{-6}$	4.1	0.0000021	0.98	$4.9x10^{-7}$	150	0.075	400	0.00016
Wet Storage Accidents - L-Reactor Basin									
Spent Nuclear Fuel Assembly Breach	0.16	0.0093	$4.7x10^{-9}$	0.00097	$4.9x10^{-10}$	0.14	0.00007	0.11	$4.4x10^{-8}$
Accidental Criticality	0.0031	170	0.000085	120	0.000060	3,000	1.5	14,000	0.0056
Aircraft Crash	$1x10^{-6}$	4.2	0.0000021	2.6	0.0000013	93	0.047	70	0.000028

[a] *New Dry Storage Facility*

Table 4-20 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory

		Consequences							
		MEI		*NPAI*		*Population*		*Worker*	
	Frequency (per yr)	*(mrem)*	*(LCF)*	*(mrem)*	*(LCF)*	*(person-rem)*	*(LCF)*	*(mrem)*	*(LCF)*
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	1.3	$6.5x10^{-7}$	0.67	$3.4x10^{-7}$	15	0.0075	28	0.000011
Dropped Spent Nuclear Fuel Cask	0.0001	0.074	$3.7x10^{-8}$	0.0033	$1.7x10^{-9}$	0.83	0.00042	0.12	$4.8x10^{-8}$
Aircraft Crash w/Fire	$1 x 10^{-6}$	180	0.00009	2.9	0.0000015	2,000	1.0	120	0.000048
Wet Storage Accidents									
Spent Nuclear Fuel Assembly Breach	0.16	0.0016	$8.0x10^{-10}$	0.0036	$1.8x10^{-9}$	0.43	0.00022	0.14	$5.6x10^{-8}$
Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1800	0.00072
Aircraft Crash	$1 x 10^{-6}$	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016

[a] *New Dry Storage Facility at IFSF/CPP-749*

Table 4-21 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	3.0	0.0000015	0.57	2.9×10^{-7}	42	0.021	50	0.000020
Dropped Spent Nuclear Fuel Cask	0.0001	0.26	1.3×10^{-7}	0.0085	4.3×10^{-9}	3.0	0.0015	0.22	8.8×10^{-8}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF									
Spent Nuclear Fuel Assembly Breach[c]	0.16	4.7	0.0000024	2.1	0.0000011	46	0.023	0.99	4.0×10^{-7}
Dropped Spent Nuclear Fuel Cask[c]	0.0001	0.2	1.0×10^{-7}	0.032	1.6×10^{-8}	3.2	0.0016	0.0049	2.0×10^{-9}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack, so workers would receive low doses.*

NA = Not applicable

Table 4-22 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	22	0.000011	42	0.000021	55	0.028	140	0.000056
Dropped Spent Nuclear Fuel Cask	0.0001	1.4	7.0×10^{-7}	0.18	9.0×10^{-8}	15	0.0075	0.61	2.4×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	2300	0.0012	180	0.000090	2900	1.5	610	0.00024

[a] *New Dry Storage Facility*

Table 4-23 Frequency and Consequences of Accidents at the Nevada Test Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	1.7	8.5×10^{-7}	0.31	1.6×10^{-7}	1.5	0.00075	20	0.0000080
Dropped Spent Nuclear Fuel Cask	0.0001	0.11	5.5×10^{-8}	0.0014	7.0×10^{-10}	0.40	0.00020	0.089	3.6×10^{-8}
Aircraft Crash w/Fire	1×10^{-6}	180	0.000090	1.2	6.0×10^{-7}	250	0.13	87	0.000035

[a] *E-MAD and New Dry Storage Facility*

The analyses were performed for a generic dry storage at the five potential foreign research reactor spent nuclear fuel management sites, as well as for site-specific locations (i.e., FMEF at the Hanford Site, E-MAD at the Nevada Test Site, L-Reactor Basin and RBOF at the Savannah River Site).

Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. These annual risk estimates are presented in Tables 4-24 through 4-28.

Table 4-24 Annual Risks of Accidents at the Savannah River Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.9×10^{-8}	5.5×10^{-9}	0.00075	0.0000018
Dropped Spent Nuclear Fuel Cask	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	1.1×10^{-11}
Aircraft Crash w/Fire	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.8×10^{-11}
Wet Storage Accidents at RBOF				
Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
Accidental Criticality	2.0×10^{-7}	7.0×10^{-8}	0.0074	0.000020
Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
Wet Storage Accidents at L-Reactor Basin				
Spent Nuclear Fuel Assembly Breach	7.4×10^{-10}	8.0×10^{-11}	0.000011	7.1×10^{-9}
Accidental Criticality	2.6×10^{-7}	1.9×10^{-7}	0.0047	0.000017
Aircraft Crash	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	2.8×10^{-11}

[a] *New Dry Storage Facility.*

Table 4-25 Annual Risks of Accidents at the Idaho National Engineering Laboratory

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.1×10^{-7}	5.5×10^{-8}	0.0012	0.0000018
Dropped Spent Nuclear Fuel Cask	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	4.8×10^{-12}
Aircraft Crash w\Fire	9.0×10^{-11}	1.5×10^{-12}	0.0000010	4.8×10^{-11}
Wet Storage Accidents[b]				
Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *FAST Facility*

The highest annual MEI or NPAI risk among the potential Phase 1 sites (Tables 4-24 and 4-25) is 2.6×10^{-7} LCF per year, which is the annual risk to the MEI from accidental criticality at L-Reactor Basin. Assuming some foreign research reactor spent nuclear fuel is stored in RBOF for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this MEI risk would be 0.0000034 LCF. The highest annual MEI or NPAI risk due to dry storage facility accidents at the potential Phase 2 sites (Tables 4-24 through 4-28) is 0.0000034 LCF per year, which is the annual risk to the NPAI from an assembly breach accident during handling at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this MEI/NPAI risk would be 0.000010 LCF. This is higher than any other

Table 4-26 Annual Risks of Accidents at the Hanford Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	2.4×10^{-7}	4.6×10^{-8}	0.0034	0.0000032
Dropped Spent Nuclear Fuel Cask	1.3×10^{-11}	4.3×10^{-13}	1.5×10^{-7}	8.8×10^{-12}
Aircraft Crash w\Fire[b]	NA	NA	NA	NA
Dry Storage Accidents at FMEF				
Spent Nuclear Fuel Assembly Breach[c]	3.7×10^{-7}	1.7×10^{-7}	0.0037	6.4×10^{-8}
Dropped Spent Nuclear Fuel Cask[c]	8.0×10^{-12}	1.6×10^{-12}	1.6×10^{-7}	2.5×10^{-13}
Aircraft Crash with Fire[b]	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack.*

NA = Not applicable

Table 4-27 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
Aircraft Crash w\Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table 4-28 Annual Risks of Accidents at the Nevada Test Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.4×10^{-7}	2.5×10^{-8}	0.00012	0.0000013
Dropped Spent Nuclear Fuel Cask	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	3.6×10^{-12}
Aircraft Crash w\Fire	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	3.5×10^{-11}

[a] *E-MAD and New Dry Storage Facility*

combination of Phase 2 annual accident risks and associated durations. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.000010 LCF for the maximum MEI risk due to accidents. This means that the MEI has one chance in one hundred thousand of incurring an LCF due to accidents.

The highest annual population risk among the potential Phase 1 sites (Tables 4-24 and 4-25) is 0.0074 LCF per year, which is the annual population risk from an accidental criticality at RBOF. Assuming some foreign research reactor spent nuclear fuel is stored in RBOF for the entire 10 years of Phase 1, plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be 0.096 LCF. The highest annual population risk due to dry storage facility accidents at the potential Phase 2 sites (Tables 4-24 through 4-28) is 0.0044 LCF per year, which is the annual risk to the public from assembly breach accidents during handling at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.013 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations. Adding the Phase 1 and Phase 2 population risks yields 0.11 LCF for the total population risk due to accidents.

Impacts of Accidents on Close-in Workers

An evaluation has been made of the impacts to close-in workers involved in spent fuel handling and management operations. This evaluation focuses on the radiological consequences of the accident. Clearly, a limited number of fatalities could occur which would be related to spent nuclear fuel handling only in an indirect or secondary manner (e.g., the worker who happened to be in the facility might be killed due to an aircraft crash).

Wet Storage Accidents

Fuel Assembly Breach in Wet Storage: No fatalities of nearby workers would be expected due to radiological consequences. This is because the release of the radionuclides would be underwater. Attenuation by the water would occur for most of the release products; release of the noble gases from the pool would, however, cause a direct radiation exposure to workers in the area. If radiation is released from the surface of the water pool, radiation alarms would sound, prompting evacuation of nearby workers.

Dropped Fuel Cask in Wet Storage: No fatalities would be expected due to the radiological consequences of this accident. The operation of the crane is done using remote controls, so workers are not likely to be in the direct vicinity of the dropped cask.

Accidental Criticality: The accidental criticality could occur at a minimum of 3 m (10 ft) underwater. Based on the shielding provided by the water pool, it is likely that no fatalities would occur. Nearby workers would likely receive appreciable radiation exposures.

Aircraft Crash into the Water Pool: No fatalities to nearby workers would be expected due to radiological consequences. An aircraft crash into the water pool would prompt nearby workers not affected by the crash to evacuate the area immediately. The release of radiation products would be underwater, allowing sufficient time for evacuation before radiation products would reach the surface.

Dry Storage Accidents

Fuel Assembly Breach: Cropping of the fuel assembly would occur in a dry cell. Any release that would occur due to inadvertent cutting into the fuel would be confined to the storage cell, where the exhaust is away from the workers. No fatalities to nearby workers would be expected due to this scenario.

Dropped Fuel Cask: No fatalities would be expected due to the radiological consequences of this accident. The operation of the crane is done using remote controls, so workers are not likely to be in the direct vicinity of the dropped cask. In addition, the workers would promptly leave the area.

Aircraft Crash with Fire: If an aircraft crashes into a dry storage facility and catches fire, large amounts of radioactive material could be released into the atmosphere. If the facility is occupied at the time of the crash, any surviving workers could receive a substantial radiation dose from the released radioactive material.

Secondary Impacts of Accidents

Impacts of accidents on resources other than human health and safety (secondary impacts), have been addressed in Section F.4 for each management site. The general conclusion is that no measurable secondary impacts to land uses, cultural resources, water quality, ecological resources, national defense, or local economies are expected from the postulated accidents involving foreign research reactor spent nuclear fuel at the management sites.

4.2.4.2 Topics Not Discussed in Detail

This section summarizes the potential impacts for the environmental topics not covered in Section 4.2.4.1, namely land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, noise, utilities and energy, and waste management. The detailed analysis of these topics presented in Appendix F, Section F.4 showed that none of these topics clearly differentiated among the potential foreign research reactor spent nuclear fuel management sites nor had any major environmental impact. The discussion of each topic generally concentrates on management sites and alternatives that have the largest estimated impacts, and demonstrates that the environmental impacts for that topic are not of sufficient magnitude to be given strong consideration in the decision making process.

4.2.4.2.1 Land Use

The basic implementation of Management Alternative 1 would only result in minor land use impacts at any of the potential foreign research reactor spent nuclear fuel management sites. The largest land use impact would be 16 ha (40 acres) at the Oak Ridge Reservation to construct a new dry storage facility. This represents less than 0.1 percent of the total size of the Oak Ridge Reservation. A description of the land use impacts at the other potential foreign research reactor spent nuclear fuel management sites is contained in Appendix F.4. For all of the potential foreign research reactor spent nuclear fuel management sites, new foreign research reactor spent nuclear fuel storage facilities would be built on land previously disturbed or designated for industrial use. No additional land outside of the existing sites would be required for foreign research reactor spent nuclear fuel management. It should be noted that land use and other environmental impacts associated with the construction activities would be minimal, under the implementation alternatives that use refurbishment of existing facilities for interim storage (i.e., BNFP at the Savannah River Site and E-MAD at the Nevada Testing Site). All environmental impacts from the refurbishment and operation of these facilities would be bounded by the impacts associated with the construction and operation of new generic storage facilities. Land use impacts are discussed in more detail in Appendix F, Section F.4.

4.2.4.2.2 Socioeconomics

The basic implementation of Management Alternative 1 would only result in minor socioeconomic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Socioeconomic impacts are defined for purposes of this analysis in terms of direct effects, which include changes in site employment and expenditures from foreign research reactor spent nuclear fuel-related construction and operation and indirect effects, such as changes that result from regional purchases, nonpayroll expenditures, and payroll spending by site employees.

No construction personnel would be needed for existing facilities, and not more than 240 workers per year (peak) would be needed to build a new dry storage facility. The annual staffing requirements for operations would be about 30 and 8 full-time employees during receipt and storage, respectively, for a new dry storage facility. This would represent 0.15 to 0.9 percent of the existing work force at any of the potential foreign research reactor spent nuclear fuel management sites. No new hiring would be expected because most positions would be filled by reassignments of the existing work force. Even if all operational positions were filled by new hires, this would represent about an even smaller increase in regional employment. The secondary effects would be even lower.

4.2.4.2.3 Cultural Resources

The basic implementation of Management Alternative 1 would only result in minor cultural impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Cultural, archaeological, historic, and architectural resources are defined as prehistoric and historic sites, districts, structures, and evidence of human use that are considered to be important to a culture, subculture, or a community for scientific, traditional, religious, or other reasons.

Although most of the potential foreign research reactor spent nuclear fuel management sites contain areas of archaeological, cultural, or historical interest, little or no direct impacts on cultural resources would be expected because of the location of the foreign research reactor spent nuclear fuel storage facilities. Specific site surveys have not been completed; however, based on existing information, no known cultural resources would be affected by construction or operation of foreign research reactor spent nuclear fuel facilities. Prior to construction, specific site surveys would be conducted. In the event that cultural resources were encountered during construction, the State Historic Preservation Officer would be contacted immediately. Similarly, Tribal leaders would be notified if any Native American resources were found.

4.2.4.2.4 Aesthetic and Scenic Resources

The basic implementation of Management Alternative 1 would only result in minor impacts to aesthetic and scenic resources at any of the potential foreign research reactor spent nuclear fuel management sites. Foreign research reactor spent nuclear fuel storage facilities would be located far from public view in areas previously disturbed or designated for industrial use. Construction activities would generate fugitive dust that could temporarily affect visibility. However, best management practices would be implemented to minimize such conditions. Furthermore, facility operations would not produce emissions that would adversely impact visibility.

4.2.4.2.5 Geology

The basic implementation of Management Alternative 1 would only result in minor geologic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Except for the potential existence of gold, tungsten, and molybdenum at the Nevada Test Site, geologic resources consist only of surficial sand, gravel, or clay deposits, all of which have low economic value. Construction activities would disturb these surface deposits, but because of the large volume of these materials on the potential foreign research reactor spent nuclear fuel management sites, the impact would be expected to be small.